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UNION CARBIDE

ORNL-TM- 2893, Vol. 1

COPY NO. - 83

DATE - October 7, 1970

TOWER SHIELDING REACTOR II
DESIGN AND OPERATION REPORT: VOL. 1 - DESCRIPTION

L. B. Holland

PREFACE

Information on the Tower Shielding Reactor II is contained in the TSR-II Design and Operation Report and in the Tower Shielding Facility Manual.

The TSR-II Design and Operation Report consists of three volumes:

ORNL-TM-2893, Volume 1, "Description of the Tower Shielding Reactor II and Facility," by L. B. Holland.

ORNL-TM-2893, Volume 2, "Safety Analysis of the Tower Shielding Reactor II," by L. B. Holland and J. O. Kolb.

ORNL-TM-2893, Volume 3, "Assembly and Testing of the Tower Shielding Reactor II Control Mechanism Housing," by D. R. Ward and L. B. Holland.

The Tower Shielding Facility Procedures Manual contains current operating, maintenance, and emergency procedures; operating safety limits; descriptions of the facility, the reactor, and auxiliary equipment and records for their operation and maintenance; delineation of the administrative organization and programs for training and qualifying personnel.

The TSR-II was conceived, designed, and fabricated by ORNL personnel. It is not feasible to give credit to the many individuals who have contributed to the overall effort but some should be singled out. At the suggestion of A. M. Weinberg that a reactor having wide utility in shielding be designed, E. P. Blizard proposed that the reactor be spherical. In order to minimize flux perturbation by control rods, C. E. Clifford proposed that the control be achieved in a central nonfueled region. P. E. Oliver first suggested the control mechanism design which utilized only metal and water in the core region. The diligent efforts of the TSF staff in carrying on development studies was a fine complement to the many individuals who contributed to the design.

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1. INTRODUCTION

The TSR-II is one of four reactors that have been operated at the Tower Shielding Facility (TSF). It is mainly used for basic shielding research and has been approved[1] for operation at 100 kW. Two of the others, the TSR-I[2] and the Aircraft Shield Test Reactor[3] (ASTR), which were approved for operation at power levels of 500 kW and 1 MW, respectively, are no longer at the TSF. The fourth reactor, the TSF-SNAP reactor, is in use and has been approved[4] for operation at a power level of 10 kW. Except for a few historical facts about the Facility and the TSR-II, this volume of the TSR-II Design and Operation Report is concerned with operation up to a maximum power level of 1 MW. Core life will be limited to 3000 MWhr.

1.1. Facility History

The Tower Shielding Facility (see Fig. 1.1) at Oak Ridge National Laboratory was built in 1954 for the original purpose of enabling studies to be made of asymmetric shield configurations for the Aircraft Nuclear Propulsion Project. This research required that the reactor radiation source be located in a region free from ground or structure scattering. The facility that was developed is still being used because of its versatility for shielding studies. It consists of four 315-ft-high towers erected on the corners of a 100- by 200-ft rectangle, with the reactor suspended between two of the towers on the 200-ft side of the rectangle. The other two towers, as well as the bridges connecting both pairs of towers, are used to support other equipment such as secondary shields and detectors. Underground buildings near the towers house the control equipment and the operating crew. A pool provides the shielding needed during shield changing and reactor servicing.

The original Tower Shielding Reactor (TSR-I) was a box-shaped 500-kW MTR-type reactor; it was replaced in 1960 with the spherically symmetric TSR-II, which has been operated at power levels of up to 100 kW and for a variety of experiments, both at ground level and at elevated positions. For a period during 1958 the Aircraft Shield Test Reactor, an MTR-type reactor used for shielding research in an operating aircraft, was under test at the TSF and was operated at a power level of 1 MW.

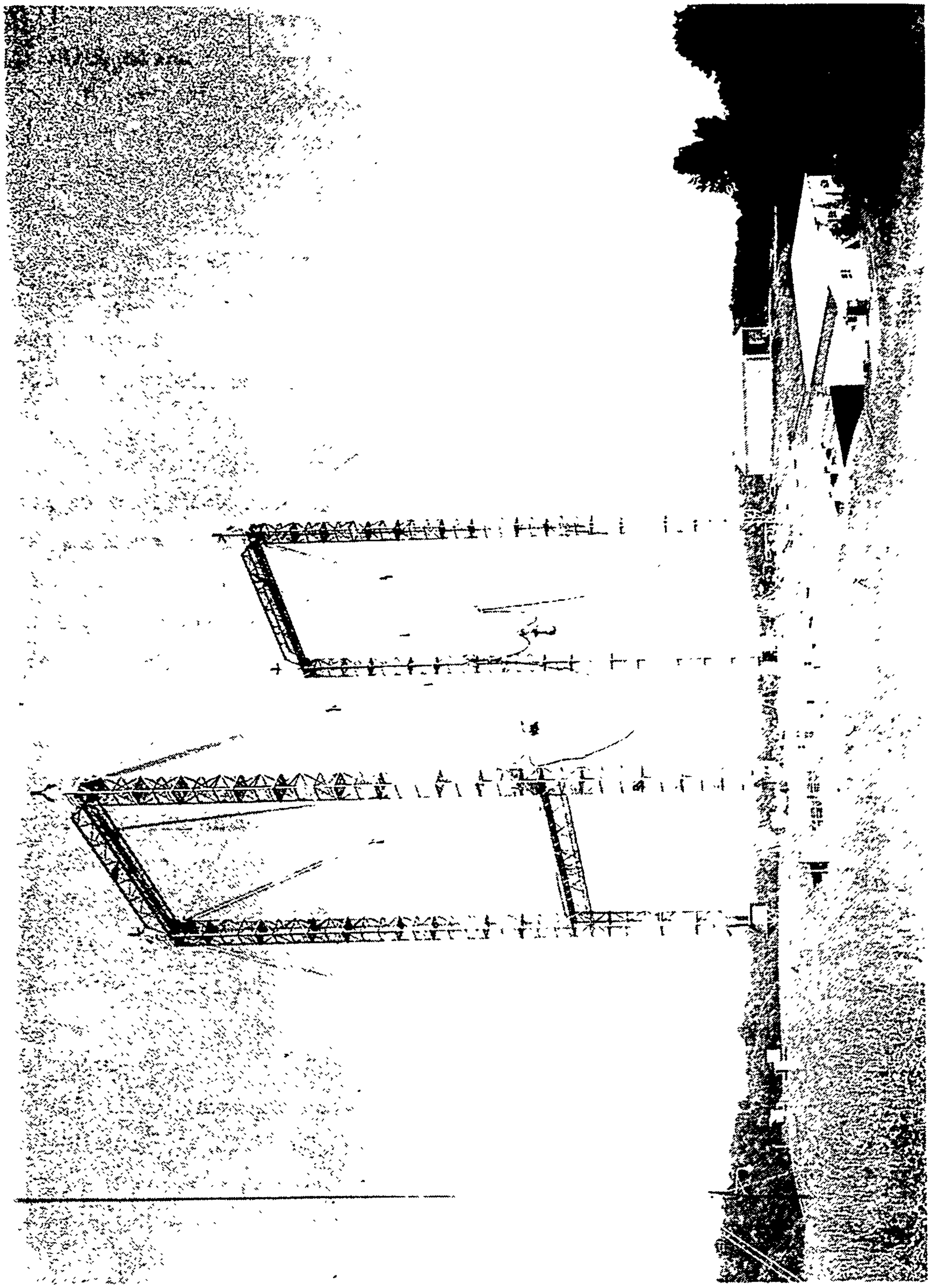

Fig. 1.1. Tower Shielding Facility.

1.2. TSR-II Operating History

The TSR-II instrumentation system was first used to operate the reactor at the Tower Shielding Facility on March 30, 1960. After several months of critical experiments, the reactor was disassembled for modifications. After reassembly the reactor was first operated with full cooling water flow on December 22, 1960. On February 6, 1961 it was operated at a maximum authorized power of 100 kW, and it has been in operation at various powers up to 100 kW since that time.

Since November 1, 1963, operation of the TSR-II has been reported semiannually.[5]

References

1. Letter dated September 29, 1959, from H. M. Roth to J. A. Swartout: "Proposed Preliminary Experiments with the Tower Shielding Reactor-II (TSR-II) at the Tower Shielding Facility (TSF)."
2. Letter dated November 4, 1954, from H. M. Roth to C. E. Larson: "Operation of TSF Reactor at 500 kW."
3. Letter dated April 28, 1958, from H. M. Roth to J. A. Swartout: "Approval for Operation of the ASTR at the TSF."
4. Letter dated June 8, 1967, from H. M. Roth to A. M. Weinberg: "Authorization for Power Operation of the TSF-SNAP."
5. L. B. Holland _et al_., Tower Shielding Reactor-II Operating Reports, unpublished reports.

2. SITE AND FACILITY DESCRIPTION

2.1. Site Location

The Tower Shielding Facility is located on a knoll with an elevation of 1069 ft 2.35 miles south-southeast of ORNL, 6 to 13 miles from the city of Oak Ridge, and 17 to 25 miles from the city of Knoxville (see Fig. 2.1). The TVA Melton Hill Dam is located 0.8 mile south of the TSF on the Clinch River, which forms a natural boundary of the restricted area. The nearest ORNL facilities, the Health Physics Research Reactor (HPRR) and the High-Flux Isotope Reactor, are over 6000 ft from the TSF and are separated from it by an offshoot of Copper Ridge and by the highest point of Copper Ridge, respectively (see Fig. 2.2). The immediate terrain on all sides of the tower structure slopes downward at the base of the towers; approximately 400 ft to the north of the towers the grade gradually rises to the top of Copper Ridge.

2.2. Personnel Barriers

Since the TSR-II is classified as an unshielded reactor, the TSF is situated within a general exclusion area (see Fig. 2.2) which is enclosed by a 6-ft-high chain-link fence topped with three strands of barbed wire (called the "perimeter fence"). Additional security is provided by an 8-ft-high chain-link fence topped with three strands of barbed wire which is located on a circle of 600-ft radius from the reactor. The TSF is separated from the other reactor, the HPRR, in the general exclusion area, by a 5-ft-high field-wire fence. Through proper operation of the reactor and continuous monitoring, the radiation levels at the perimeter fence are not allowed to exceed the radiation protection standards for uncontrolled areas.

2.3. Underground Buildings

Two reinforced-concrete underground buildings, with inside dimensions of 59 x 30 x 10 ft and 115 x 30 x 10 ft, adjacent to and north of the towers (see Fig. 2.3) provide a shielded working area for personnel during reactor operation. The smaller building (7704) is used primarily as a service and shop area and is connected to the larger building (7702) by an 8-ft-wide walkway. The larger building contains the reactor controls,

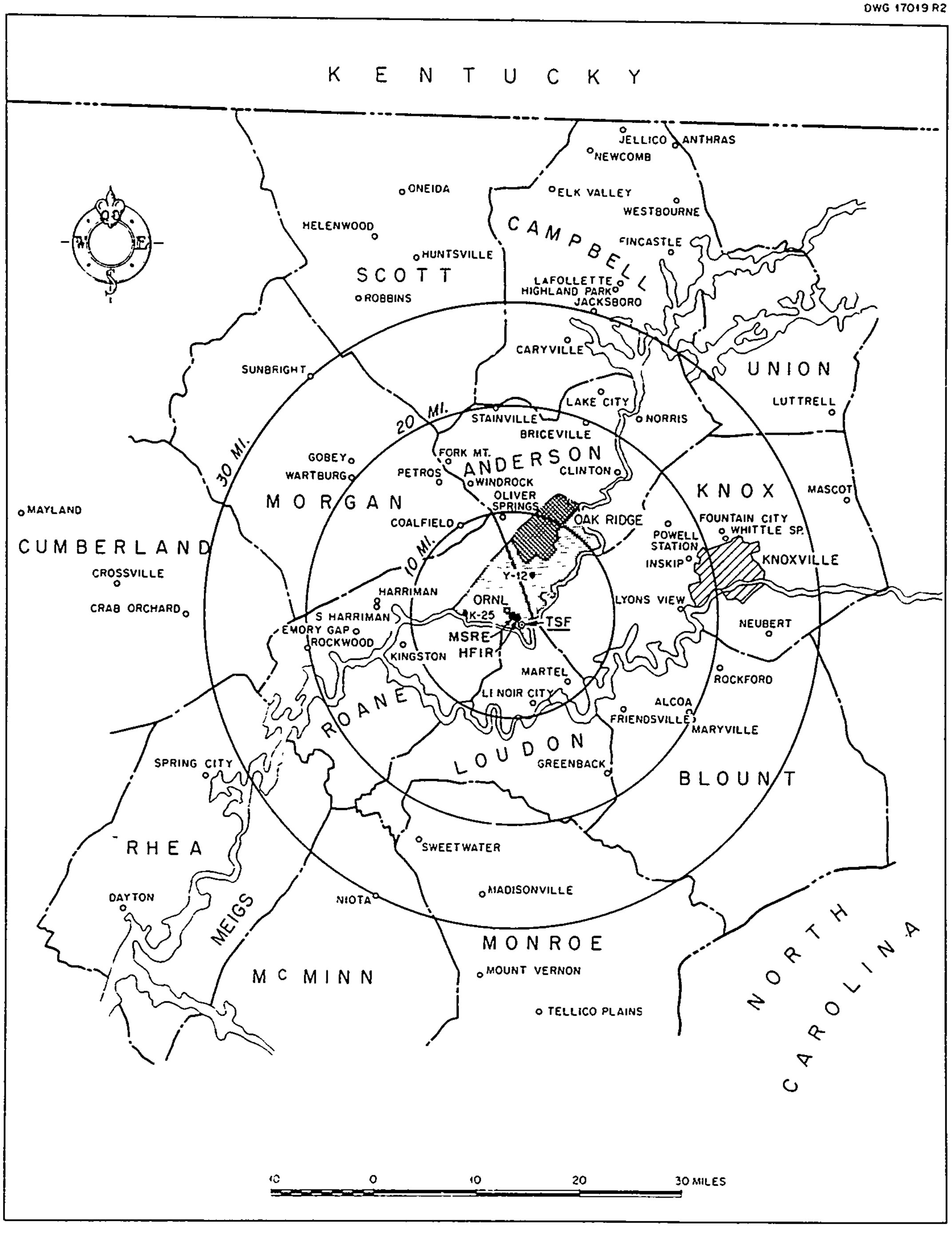

Fig. 2.1. Map of cities and counties surrounding TSF area.

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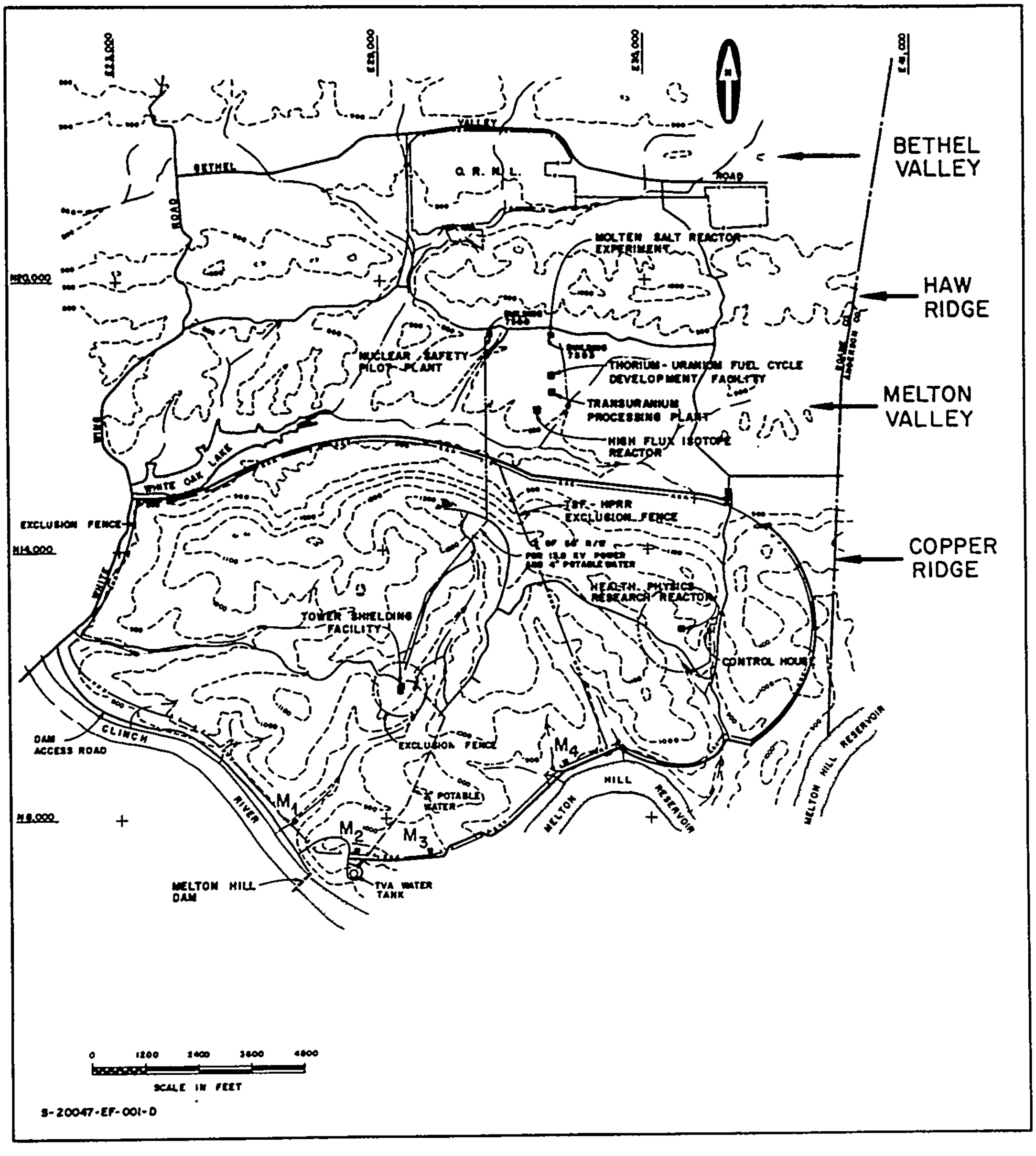

Fig. 2.2. Area topography.

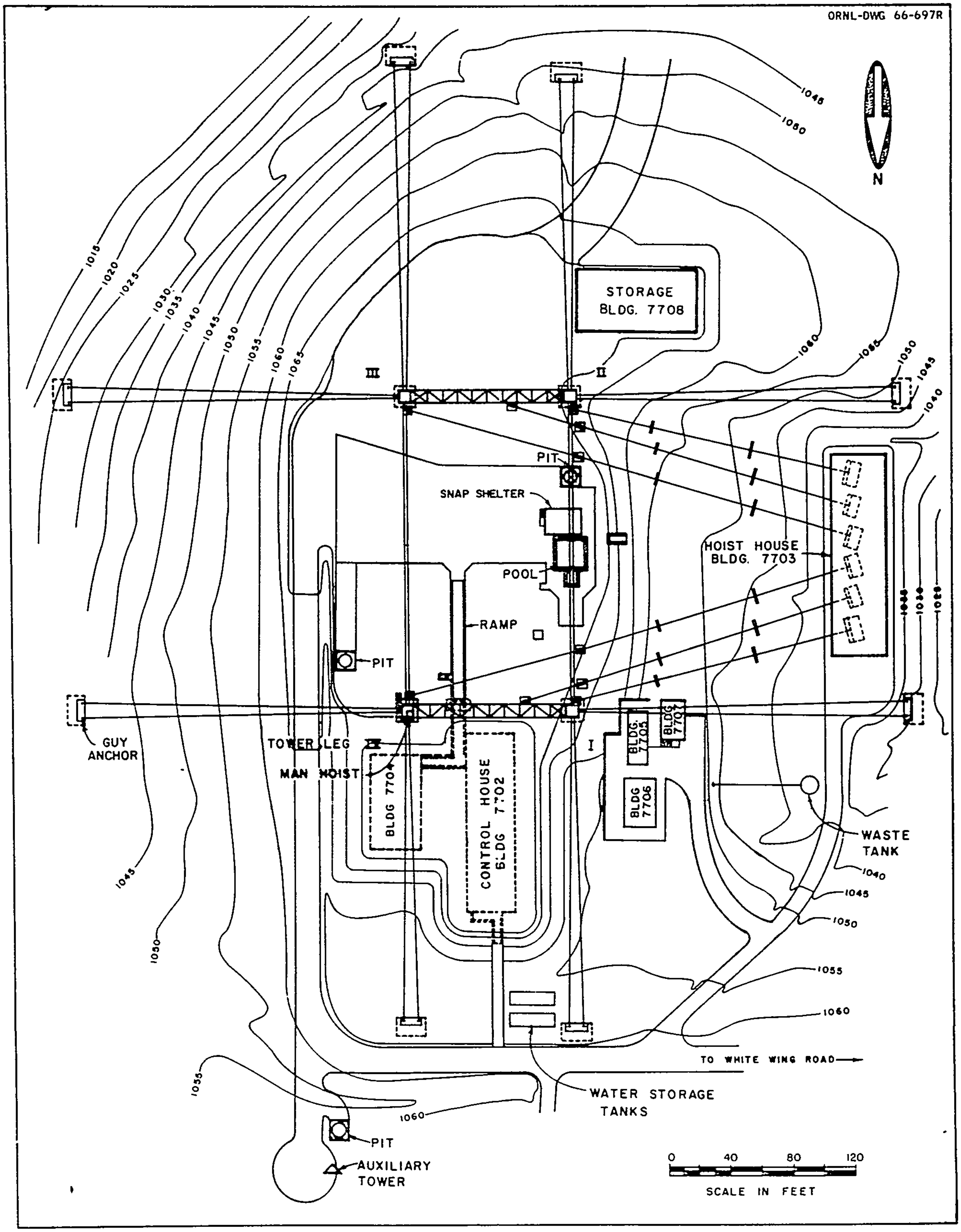

Fig. 2.3. Site plan, TSF, ORNL.

data-collecting facility, counting room, and offices (see Fig. 2.4). The buildings are shielded against radiation by an 18-in.-thick concrete roof covered with 3 1/2 ft of earth. To ensure that air entering the underground buildings is free from particulate matter and that there is a slightly positive air pressure inside, blowers force air through High Efficiency Particulate Air and activated charcoal filters into each building, and the air leaves the buildings through a common duct which has a weighted damper.

2.4. Tower and Facility Layout

2.4.1. Tower Description

The tower structure is a braced and guyed steel frame forming a 100- by 200-ft rectangle, with a leg placed at each of the four corners (Figs. 2.3 and 2.5). Each leg is 9 ft square and 315 ft high, and terminates at the lower end in an inverted truncated pyramid. Since the objective of the design was to minimize the scattering of radiation by structural material, the unit weight of the steel in the tower structure was kept below 400 lb/ft. Each pair of legs is joined at the top by a horizontal truss-type bridge running east and west as shown in the three-dimensional view of Fig. 2.5. Maintenance access is provided by a bridge between legs I and IV of the north tower at the 100-ft level; the bridge is reached by an elevator in leg IV. In 1963 the design calculations of the towers and the reactor support structure were reviewed[1] and found to be in conformity with the "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings" of the American Institute of Steel Construction (AISC) for all operating conditions.

2.4.2. Guy Cables

A pair of galvanized plow-steel cables (or sister guys) 2 in. in diameter stretches between leg I of the north tower and leg II of the south tower, and another pair stretches between leg IV of the north tower and leg III of the south tower. Each cable of the pair was designed to be held in 40,000-lb tension by two similarly constructed pairs of guys which extend from the top of the towers to ground anchors in the north-south direction. The towers are guyed in the east-west direction by eight

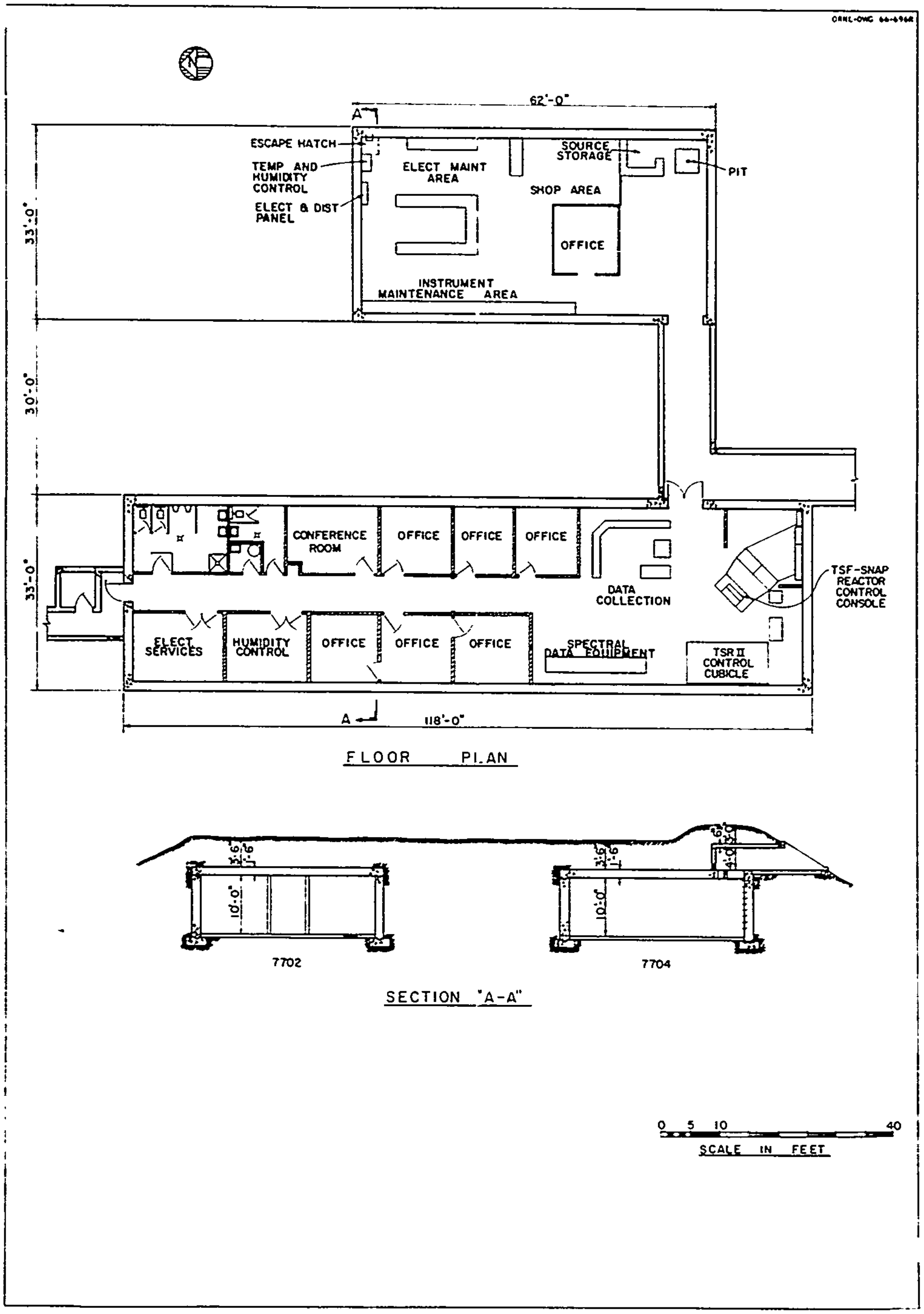

Fig. 2.4. Control house.

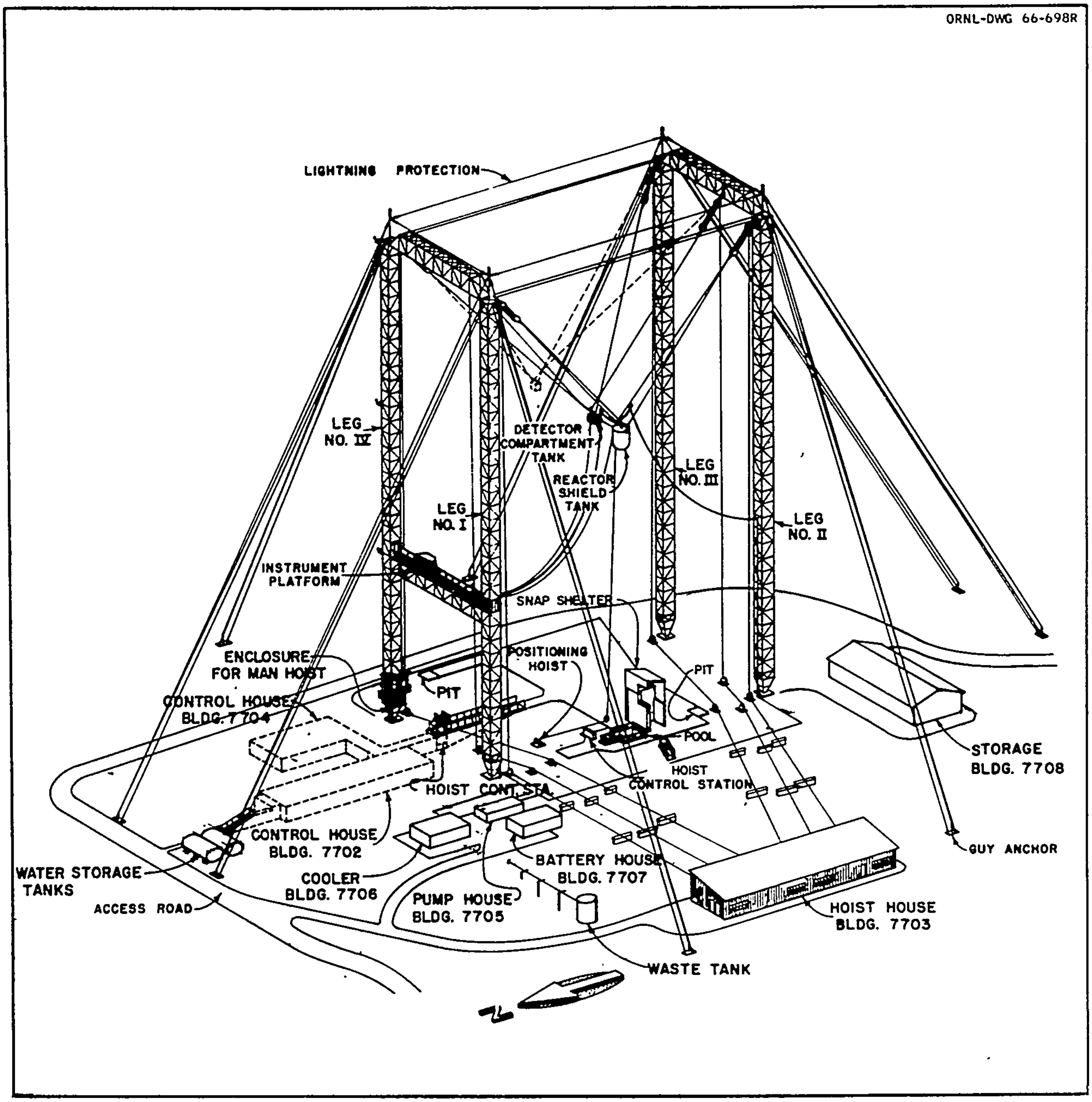

Fig. 2.5. Trimetric of TSF.

inclined guy cables. Each cable was originally set at 76,000-lb tension, and makes an angle of 32° with the towers.

2.4.3. Wind Loading

Although the towers were designed to deflect no more than 11 in. during a 105-mph wind with the maximum loads at altitude, the reactor is not elevated for operation when winds exceed 40 mph, and if it is at the maximum elevated position when a thunderstorm approaches, it is lowered. (The factor of safety of the towers for a 105 mph wind was calculated to be 1.7.) To ensure that operations are not conducted in dangerous wind conditions, daily communication is maintained with the U. S. Weather Bureau, and local observations are made with an anemometer and a wind-direction indicator. In addition, the TSF is equipped with weather radar to track local thunderstorms.

2.4.4. Lightning Protection System

The Tower structure is completely shielded by means of a wire grounding net. As shown in Fig. 2.5, a copper-clad steel-strand shielding wire is mounted on porcelain insulators to form a rectangle at a 5-ft minimum above the entire structure. Shielding wire also extends from the top of each tower to the ground to protect the inclined guys. The steel towers, the inclined guy wires, and the grounding system are connected to a buried counterpoise. The resistance to the ground of the above-ground grid system is between 1 and 3 ohms.

2.5. Hoisting Equipment

The equipment for hoisting the reactor and for hoisting and positioning other loads consists of six major hoists, designed to be operated in pairs or singly, with the hoist drums and motors located in a hoist house to the west of the towers. The hoisting lines on the necessary ground sheaves, idlers, etc., extend along the ground from the hoist house to the base of the towers and then vertically to fixed multipart sheave blocks near the top of the columns or on the bridges atop the towers, and then down to multipart traveling blocks. The reactor traveling blocks are connected to the loads by pairs of 2-in.-diam cables; the other blocks are connected to their loads by single 2-in.-diam cables. Two of the hoists, which can lift 55 tons to an altitude of 200 ft, are used to move the reactor and shield in the plane of the west tower legs (I and II). Another pair of hoists can raise a load of 40 tons to the same altitude in a plane 35 ft east of the plane of legs I and II. The third pair of hoists can lift 30 tons in the plane of the east tower legs (IV and III) or, together with the second pair of hoists, can be connected to one 30-ton load so that the load can be positioned at various distances from the reactor. During this type of operation an auxiliary constant-tension hoist can be used to keep the separation distance between the auxiliary load and the reactor constant even if there is oscillatory motion due to winds.

There are two separate and independent hoist control stations: the local control station, which is located in a partially shielded ramp at the south entrance to the underground buildings and contains a closed-circuit television for observing the moving loads; and the remote control station, which is located adjacent to the reactor-handling pool. The hoist controls are designed to provide smooth acceleration and deceleration through five speeds and to brake automatically in case of power failure. In the event of an emergency an auxiliary battery power source can be operated from either hoist control station to release the brakes and thus lower the loads. The loads and the towers are protected from operator error by limit switches and slack-line switches. Also, moving loads can be observed by television from the reactor console. Procedures for operating the hoists are given in Chapter 6 of the TSF Manual.

2.6. Reactor Suspension System

The core assembly (see Section 3.1) is supported in an aluminum pressure vessel (see Section 3.2). For in-air operation the pressure vessel will be supported from the support platform (see Fig. 2.6), which in turn is suspended by shackles from the cables of tower legs I and II. In the center of the platform is a 4-ft-diam ball-bearing assembly, whose outer race is anchored to the platform. The reactor (and also its shield if one is used) is supported from the inner race. This permits the reactor to be rotated around its vertical axis for certain experiments. The inner bearing race extension, shield support collar, and reactor support rim are equipped with a series of discontinuous interlocking ledges so that the support platform can be used

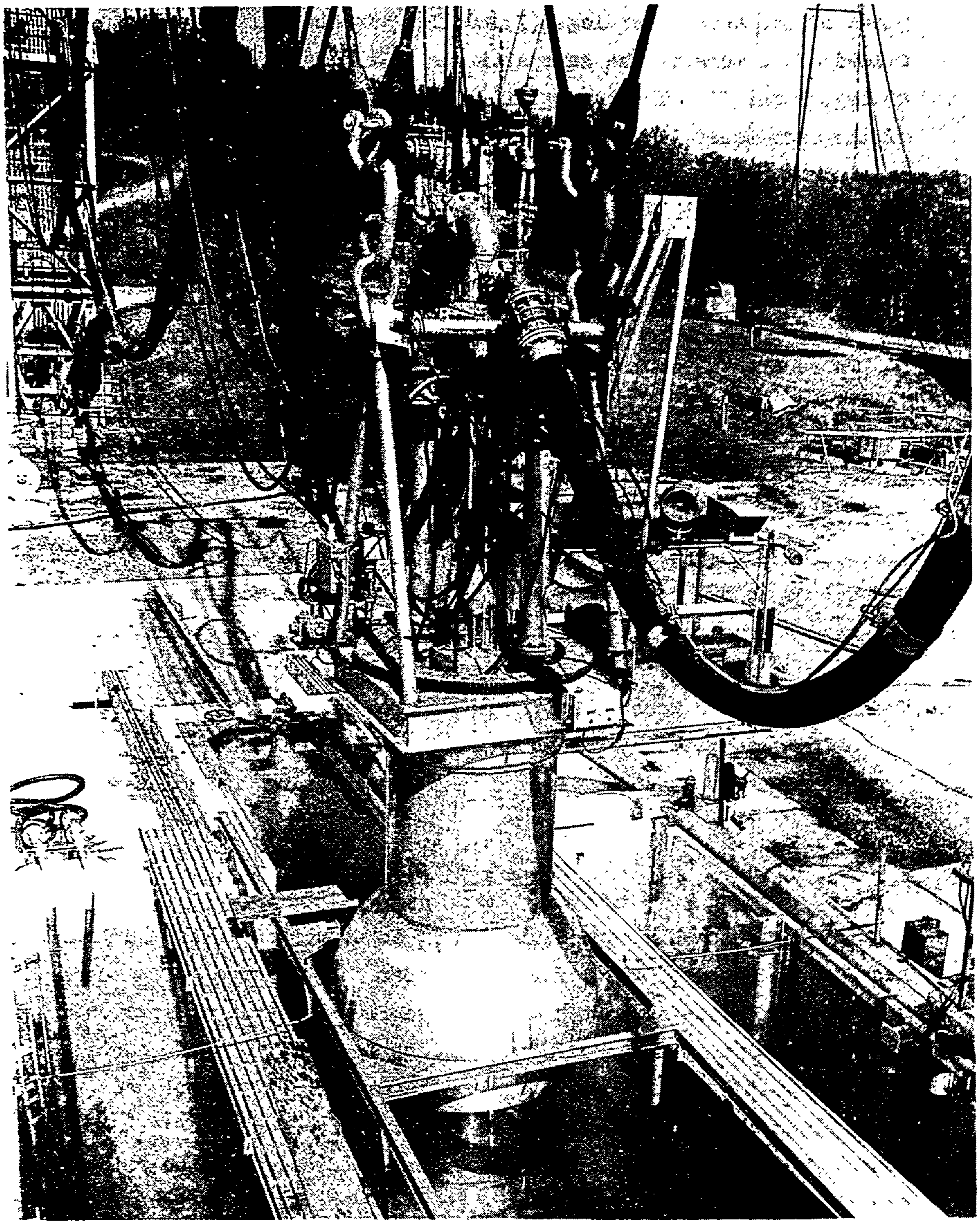

Fig. 2.6. Reactor in COOL-II shield suspended from reactor support platform.

for lifting the reactor alone or for lifting the reactor and shield together as a unit, the mode of lifting depending on the angular position of the inner bearing race. The overall arrangement of the reactor, support platform, hoses, and electrical cables suspended from the towers at two different elevations is shown in Fig. 2.7.

2.7. Inspection Procedures for the Reactor Suspension System

Inspection procedures for the reactor suspension system enable detection of points of wear damage and fatigue which would weaken the system. In particular, an eddy current technique has been developed which measures reduction in cross-sectional areas in the hoist and guy cables. A reduction of less than 2% in the cable area can be detected and a reduction in area of 4% has been noted at one point. Even so, all but one* of the cables still qualify to lift the load for which they were commissioned in 1954. If cables or other suspension system components were to deteriorate, the component would, of course, be taken out of service. If the reduction in area of a cable were 12% or more, it would be taken out of service even if the cable appeared to be sound.

2.8. Reactor- and Shield-Handling Pool

A two-section reinforced concrete pool provides shielding during the removal and storage of fuel elements and the changing of reactor shields. The pool is located midway between the west tower legs; its large section is 20 ft square and 25 ft deep, and its small section is 4 ft wide by 12 ft long by 22 ft deep (see Fig. 2.8). Some of the reactor shields have a minimum amount of structural material in the outer surface to minimize unwanted

*A hoisting cable was broken in 1963 but the suspended load was not dropped.[2] The cable was severely damaged when sensing devices failed to indicate that the cable had become slack on the drum. The cable was replaced and tested under maximum load. The slack-cable sensing system was redesigned to utilize multiple sensors independently as they are used in a reactor safety system. Also, operational checkout of the system has been improved and the frequency of checkout increased as outlined in the TSF Procedures Manual.

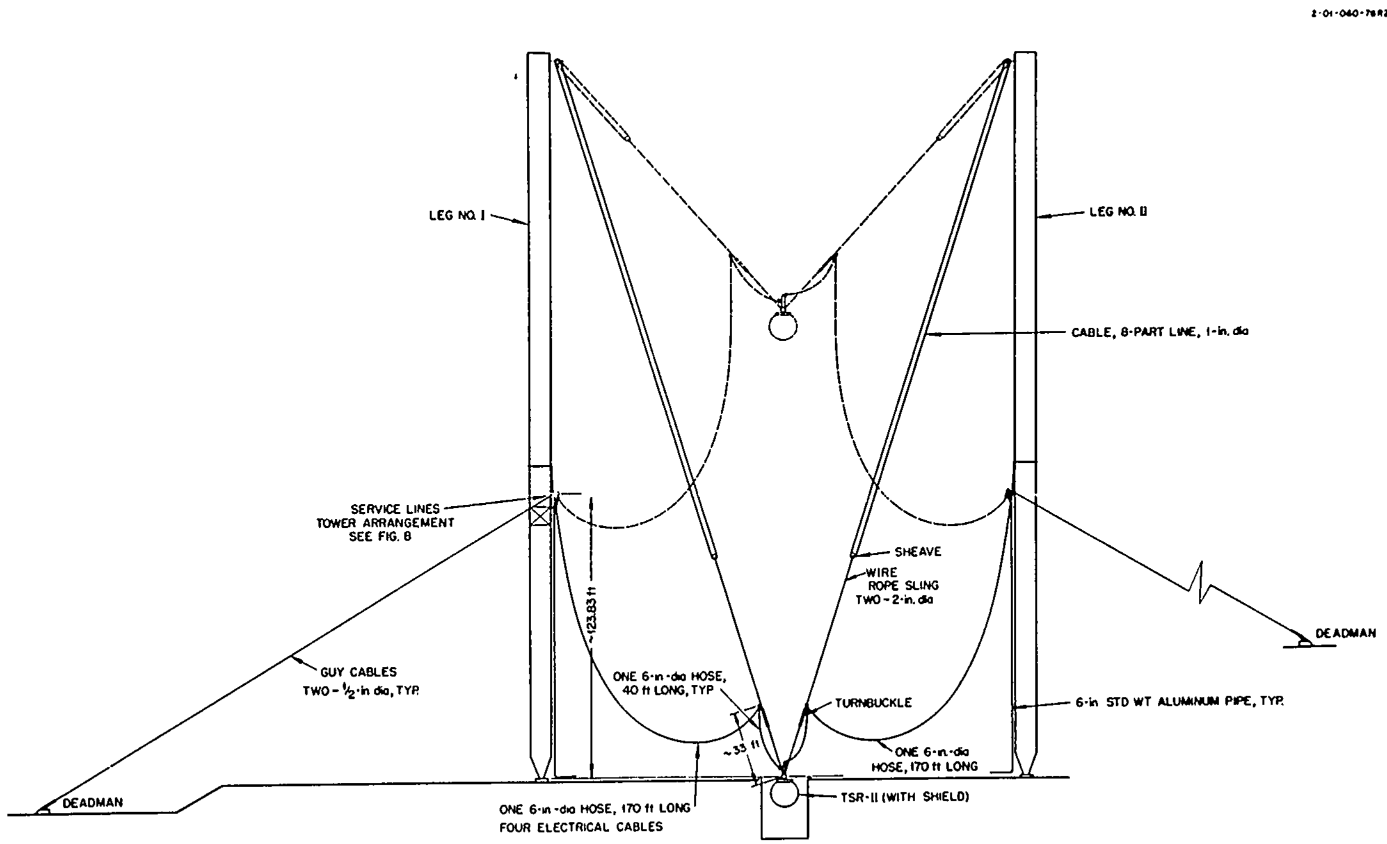

Fig. 2.7. Reactor platform suspension arrangement (TSR-II in beam shield).

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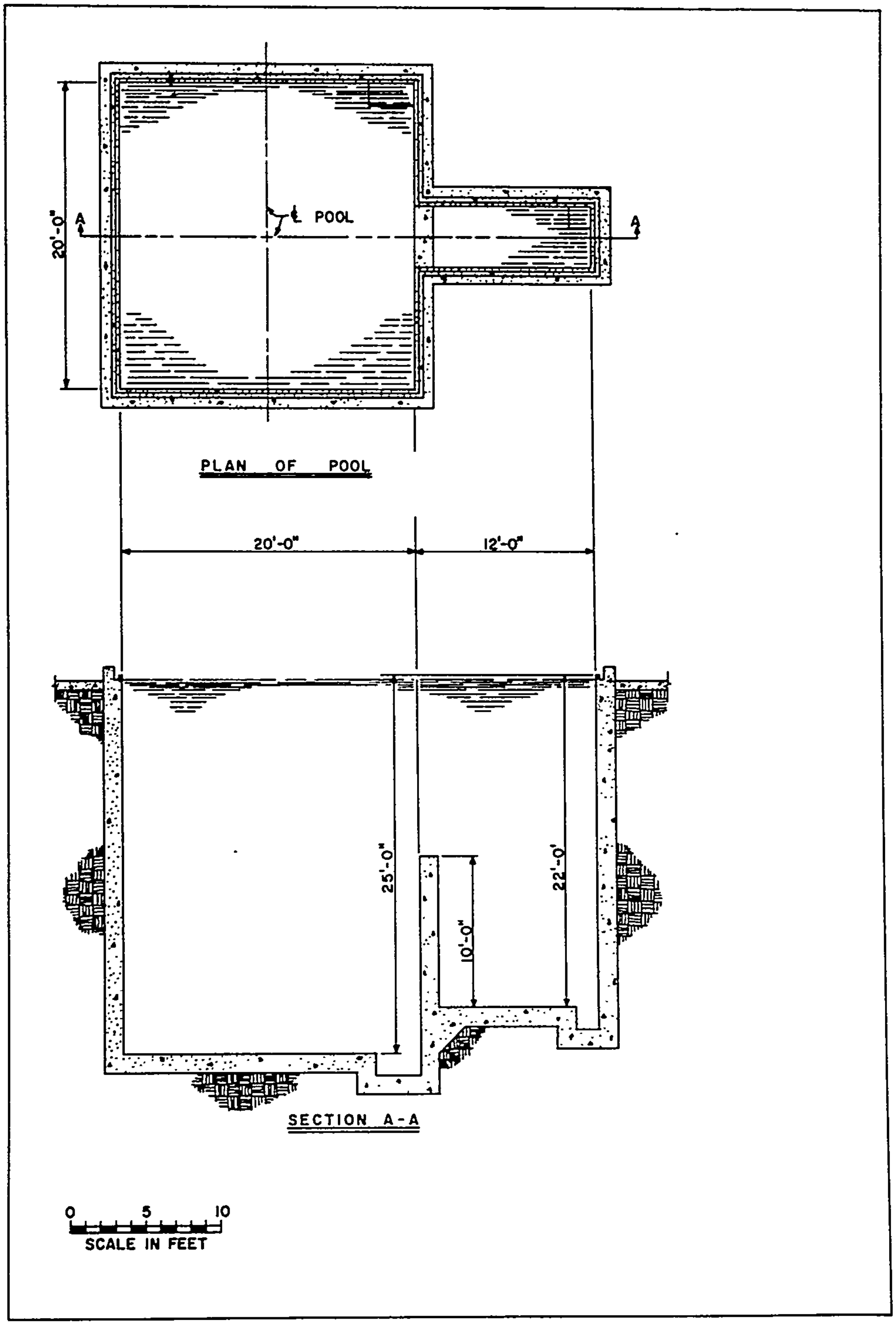

Fig. 2.8. Reactor and shield handling pool.

nuclear effects, and these shields can be supported only by "lifting collars." A guided float is installed in the pool for raising and lowering these shields (see Fig. 2.9). In order to raise the shields, the float, with the shield in place, is lowered to the bottom of the pool, the reactor is suspended above it, and then the float is raised to the reactor and the supporting collars are engaged to lock the shield in place around the reactor. The water in the pool is circulated through a system of filters to keep it clear for the above operations. Since maintenance involving the handling of fuel elements occurs so infrequently, the pool is drained and used for other purposes as noted in Section 2.9.

2.9. TSF-SNAP Installation

The TSF-SNAP reactor is a compact, beryllium-reflected, NaK-cooled, hydrided-zirconium-uranium alloy fueled reactor which is operated while suspended from a support structure. The support structure is mounted on a 2-ft-thick concrete slab which covers the south section of the drained reactor handling pool. Two other similar concrete slabs completely cover the remaining portion of the large section of the handling pool except for a 4-ft-diameter hole in the center slab. Shields and detector are placed in the drained pool to perform shielding studies. The reactor can be swung from a position directly over the center of the drained pool counter clockwise through approximately 190° into a shelter which is used as a storage and maintenance area for the TSF-SNAP reactor. The shelter which is approximately 14-ft square inside is immediately adjacent to the south side of the handling pool. The lower portion of walls of the shelter is solid concrete block for shielding and the upper portion and the roof are sheet metal.

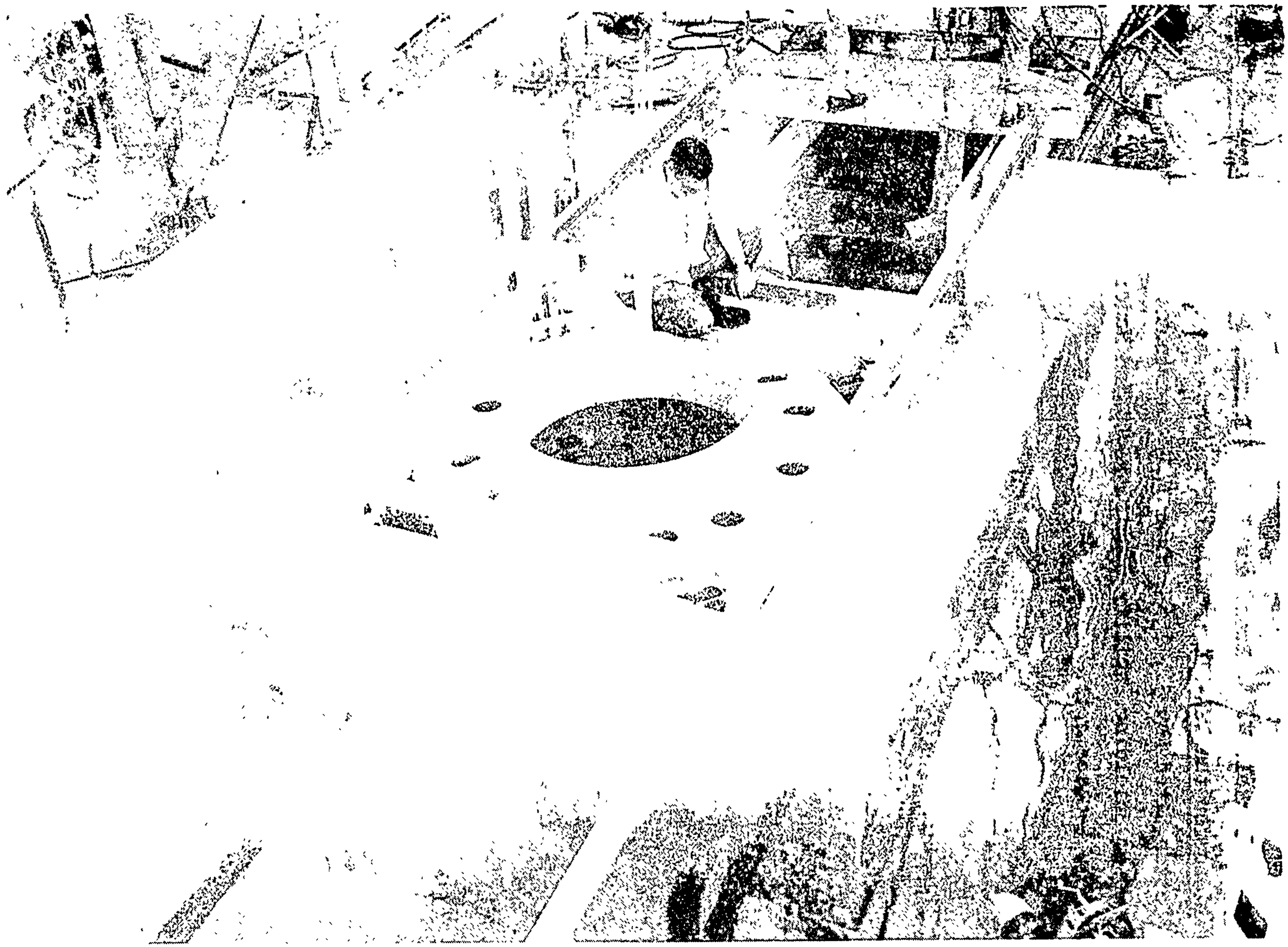

Fig. 2.9. Shield handling float.

References

1. T. W. Pickel, W. A. Bush, and B. W. Wieland, Review of Stress Calculations for the TSR-II, unpublished report (May 27, 1963).
2. Letter dated May 31, 1963, from J. A. Swartout to H. M. Roth, subject: "Tower Shielding Facility Control Mechanism Housing and Hoisting Cable Replacements."

3. REACTOR DESCRIPTION*

3.1. Description of Core Region

The TSR-II core consists of 60-mil-thick curved aluminum-clad uranium-aluminum alloy plates cooled and moderated with light water. The plates are shaped so that the assembled core is a spherical fuel annulus from which the radiation is emitted symmetrically. The neutron-absorbing shim-safety control plates** for the reactor are contained in the fuel-free region centered inside the fuel annulus. Outside the fuel annulus is a reflector region, which may contain aluminum-water, lead-boral-aluminum, or other combinations of materials as an experiment demands.

3.2. Pressure Vessel

The three regions of the TSR-II core, the internal reflector, the fuel annulus, and the other reflector, are located in the lower section of a cylindrical aluminum tank with a hemispherical bottom, as shown in Fig. 3.1. This aluminum tank is 8 ft long; its inside diameter at the hemispherical end is 37 in., which is increased to 40 in. at the open end to facilitate removal of fuel elements, shielding etc., from the tank.

3.3. Fuel Annulus

The spherical fuel annulus is 5.5 in. thick and 29 in. in outside diameter and consists of 21 fuel elements fabricated so that the fuel plates in adjacent elements join to form many concentric cylinders separated by water passages. The spherical contour of the core was accomplished by varying the lengths, widths, and radii of the fuel plates (see Fig. 3.2).

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*The TSR-II was designed by ORNL General Engineering personnel. The mechanical design calculations, which were detailed in two unpublished reports,[1,2] were based in part on the ASME Boiler and Pressure Vessel Code, 1956 edition, Section VIII, "Unfired Pressure Vessels." The design which was completed prior to the publication of the first Nuclear Code Cases[3] was reviewed in 1963.[4]

**The neutron absorbing shim-safety plates used to operate and shut down the reactor have previously been referred to as rods because of the general usage of the term.

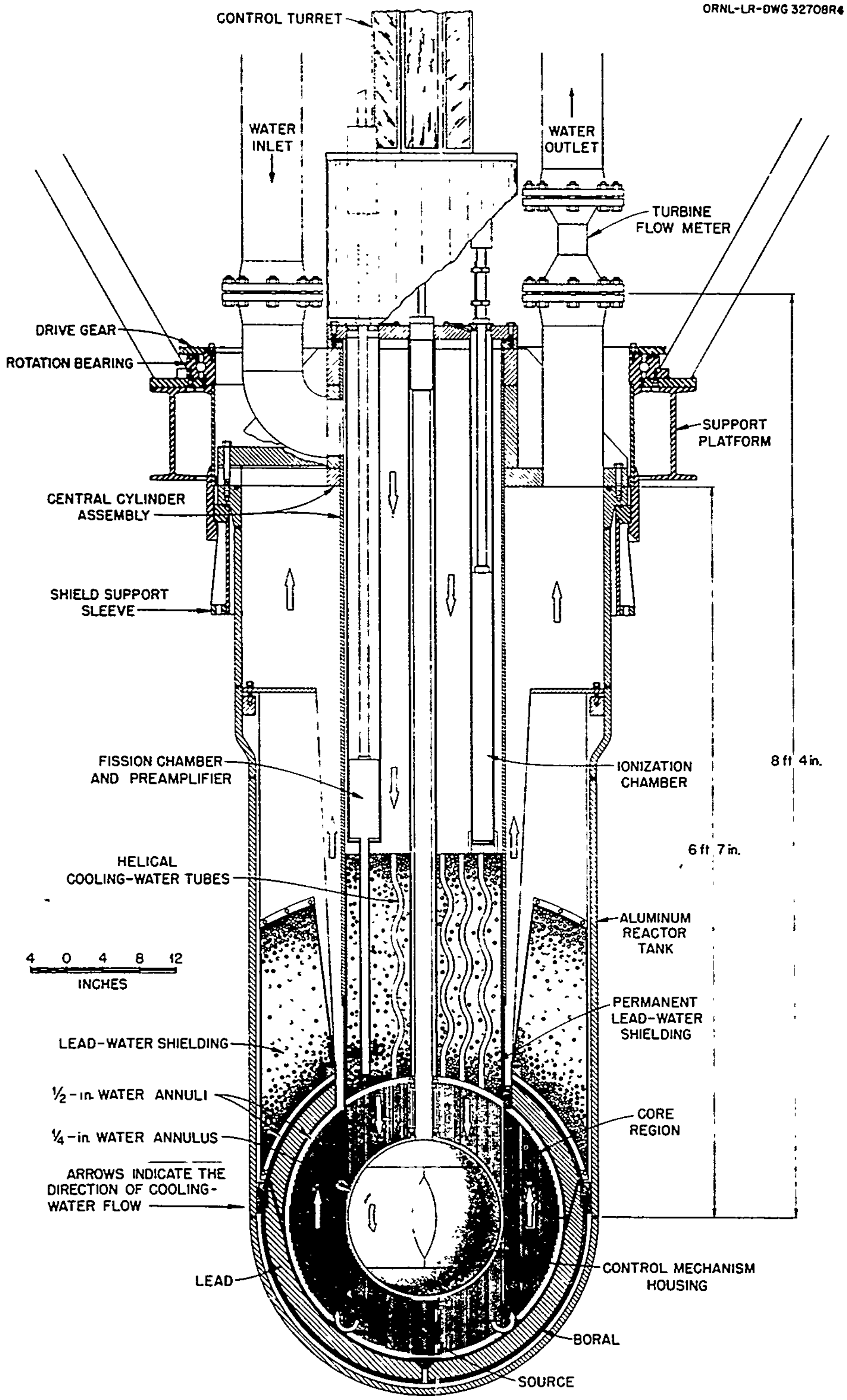

Fig. 3.1. Tower Shielding Reactor-II (vertical section).

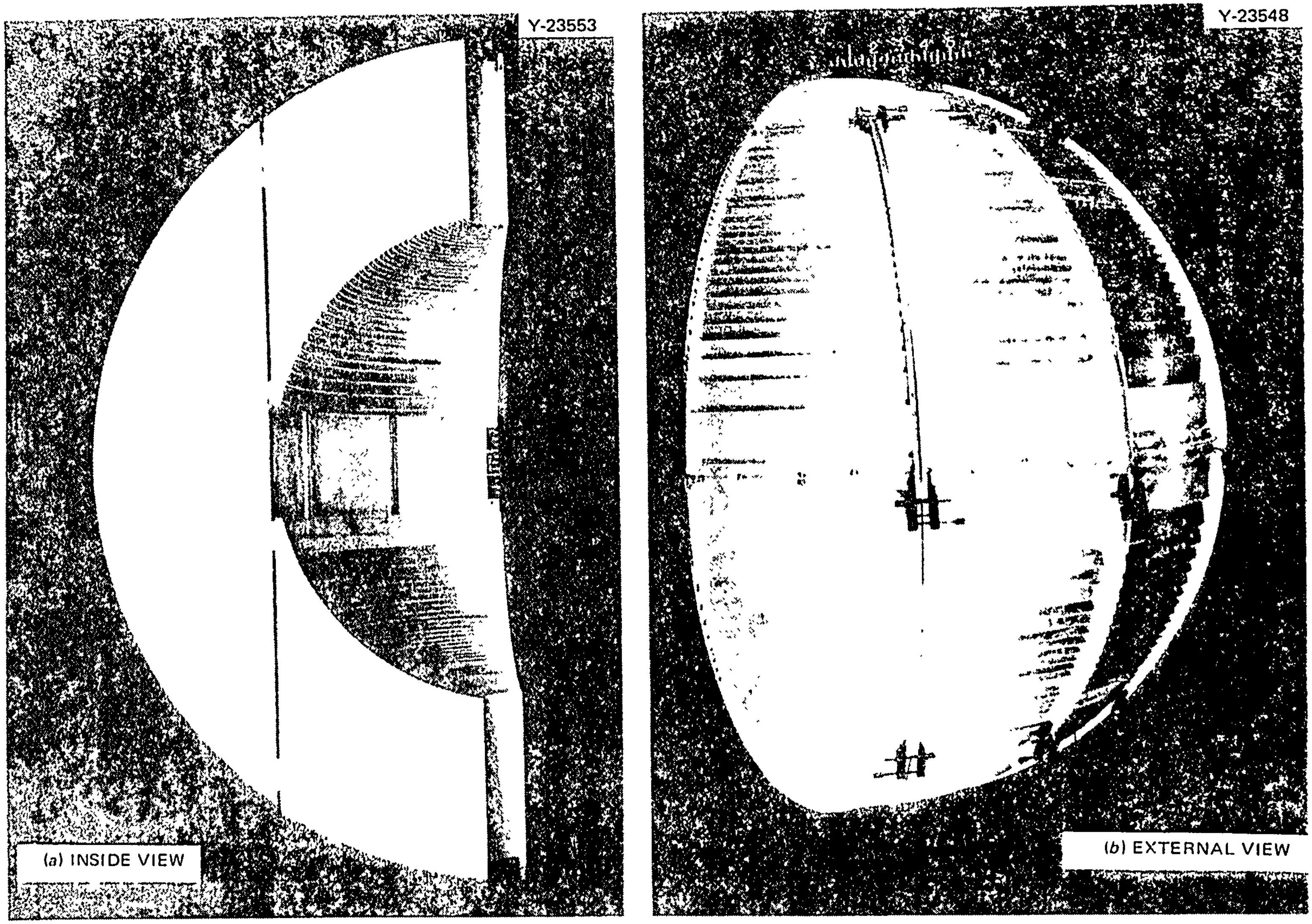

Fig. 3.2. Fuel elements assembled in a quadrant.

3.4. Fuel Elements

Each fuel plate is 0.060 in. thick* and consists of a sandwich of uranium-aluminum alloy clad in aluminum. Fuel plates are peened and welded 0.120 in. apart into aluminum side plates to form elements. Three types of elements are used: annular elements, so called because they form a cylindrical fuel annulus when assembled together; central elements, which are used in the upper and lower sections of the core; and one 3-in.-diam cylindrical "plug" element, which is centered in the lower central elements and which contains an antimony-beryllium source (also see Section 7.12). The central elements, two of which are shown in Fig. 3.2a, are contained within a bottomless aluminum cylinder positioned inside the reactor pressure vessel, and the annular elements, shown in Fig. 3.2b, are held in the region between the central cylinder and the reactor tank. There are 12 annular elements and 8 central elements (four upper and four lower). A 1/8-in.-thick fuel-loaded aluminum shell is mounted on the control mechanism housing and is located 1/4 in. inside the elements in the fuel annulus.

3.5. Control Elements and Control Mechanism Housing

The internal reflector region is almost completely filled by a 17-in.-diam aluminum sphere which houses six neutron absorbing control plates and the mechanisms for positioning them to operate and shut down the reactor (see Fig. 3.3). Four fuel-loaded, lune-shaped covers which are mounted outside the control mechanism housing form a complete spherical shell. A neutron source is permanently located in the aluminum structure (also see Section 7.12). The completely assembled sphere mounts on four blocks which are welded on the inside of the central cylinder at the horizontal reactor midplane. Each control plate is a dished, hermetically-sealed hollow plate of 1/16-in.-thick stainless steel filled with boron carbide. The total thickness of each plate is 1/2 in. Five plates, designated as shim-safety plates, are moved simultaneously relative to the fuel to operate the reactor, but each plate is independently driven toward the fuel, four outward and

*The fuel plate thickness and spacing are basically the same as those of the pool-type Bulk Shielding Reactor.

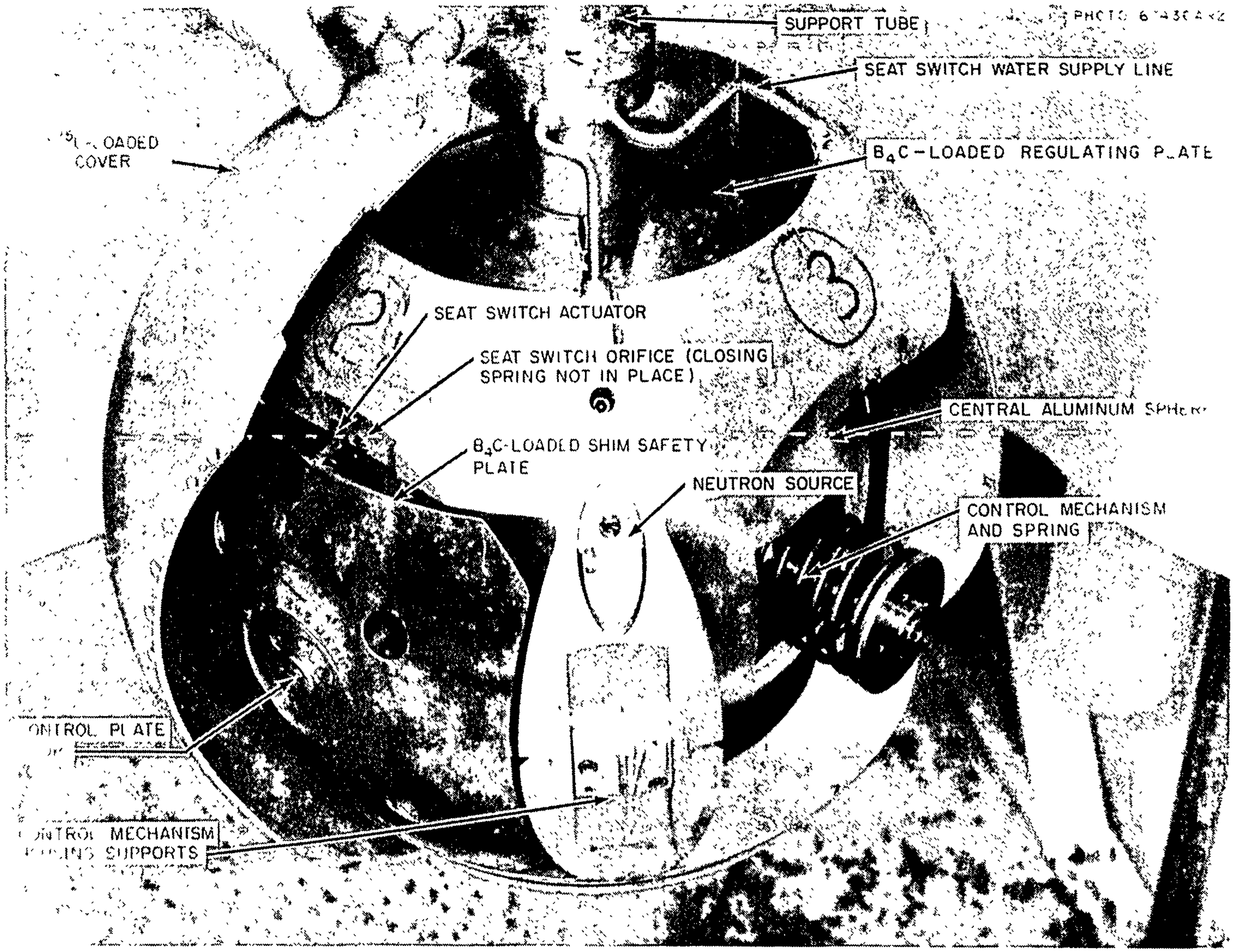

Fig. 3.3. Control mechanism housing.

one downward, to shut down the reactor when the protective system initiates the action. The sixth plate, which is designated as the regulating plate, moves vertically in the upper region of the sphere and can be servo-operated to maintain the reactor power at a constant level. All cavities within the spherical control mechanism housing are filled with water. The reactor is thus controlled by varying the thickness of water between the control plates and the inner surface of the fuel annulus. Movement of the control plates is achieved by a combination of mechanical and hydraulic forces, as described below.

3.6. Control Mechanism

In each control mechanism (Fig. 3.4) a throttling action develops a hydraulic pressure differential across a piston to balance a spring force so that a small motor can simultaneously position the shim-safety plates connected to five separate control mechanisms to operate the reactor. The interruption of the water flow producing the pressure differential permits the spring on each mechanism to independently drive its associated plate toward the fuel to shut down the reactor. The design and operation of a control mechanism are described below.

Each shim-safety plate is connected to a piston which moves approximately 1.7 in. along a guide shaft centered in a stainless steel cylinder. A spring which is held with a compressive force of 70 to 90 lb is designed to return the plate to a position adjacent to the fuel to shut down the reactor or to hold it in that position during nonoperating periods. In this position the piston is at the left end of the cylinder as in view b of Fig. 3.4.

Centered inside the guide shaft is a lead screw with a traveling nut which engages a spool through two lengthwise slots in the guide shaft. Thus the spool can be moved to any position along the guide shaft by proper rotation of the lead screw. Axial holes for water flow preclude the buildup of any pressure differential across the spool. A pair of knife edges formed by circumferential V-shaped rings on the hardened stainless steel spool are fabricated to have a few mils radial clearance with matching knife edges inside the cup-shaped end of the piston. Between the rings on the inside of the piston there are 24 radial holes for water flow.

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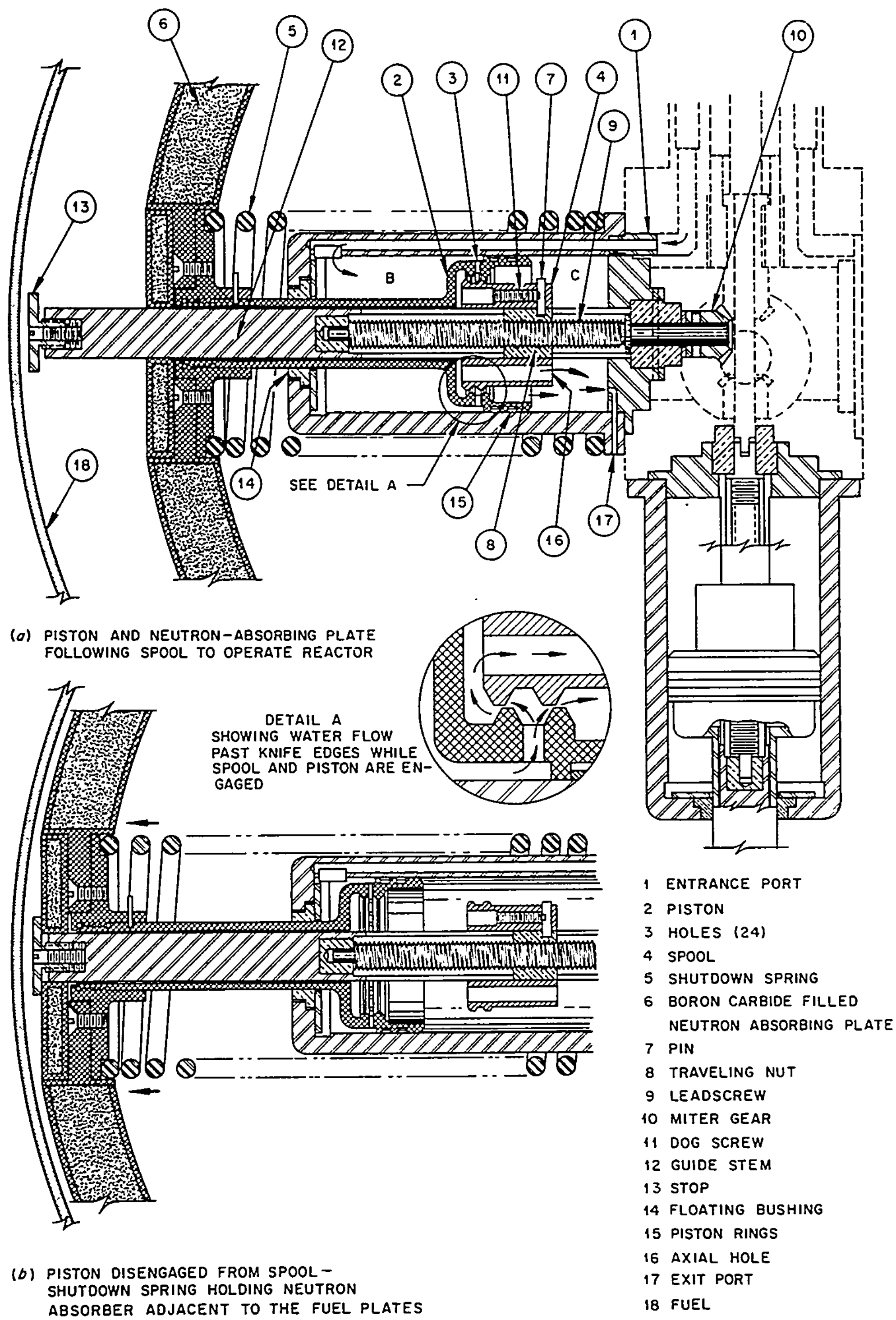

Fig. 3.4. Control mechanism.

A single 7-W ac control motor mounted in a turret atop the reactor pressure vessel is used to drive the lead screws on all five control mechanisms. Through a 64:1 speed reduction the motor rotates a vertical shaft which extends to the center of the control mechanism housing, where 1:1 miter gears drive the lead screws in the control mechanisms.

A pump located in the reactor-cooling-water pump room supplies the water flow to provide the pressure differential in the mechanism. The filtered water passes through a solenoid shutdown valve which is part of the reactor protection system (see Section 7.4). There is a shutdown valve for each control mechanism. During nonoperating periods the valve, which is located in the turret, diverts the water flow from the pump to the cooling water in the pressure vessel.

To operate the reactor the shim-safety plate must be moved to the right (as shown in Fig 3.4) which is away from the fuel and against the spring force. Since there is no physical connection between the spool with its positioning drive and the neutron absorbing plate and its piston, a pressure differential must be generated across the piston to move it against the spring force. To generate the pressure differential, the reactor protection system must be operating so that the shutdown valve directs the water flow to the entrance port of the control mechanism. The water flows through an axial hole in the cylinder wall to the volume B (see view a Fig. 3.4) to the left of the piston, then passes through the radial holes in the piston to the volume C to the right of the piston, and finally exits through ports at the right end of the cylinder. If the knife edges of the spool are not in juxtaposition with those on the piston (see view b Fig. 3.4), the water will flow through the cylinder without generating the required pressure differential. When the spool is moved to the left so that its knife edges are in juxtaposition with those of the piston (see view a and detail "A" Fig. 3.4), they throttle the water flowing through the ports in the piston so that a pressure differential is generated across the piston, with the net force moving the piston to the right until the leakage between the knife edges is such that the force on the piston due to the pressure differential just equals the spring force. Then as the spool is moved to the right, the gap between the knife edges closes which increases the differential pressure across the piston, and the increased pressure differential moves the piston to the right compressing the spring and opening the gap. Conversely as the spool is moved to the left, the gap between the knife edges opens which decreases the pressure differential and the spring forces the piston to the left to close the gap.

The pressure differential could be generated with a single pair of knife edges. The use of the second pair of knife edges, however, ensures that the only force on the spool due to the throttling action of the knife edges is radial and since it is circumferentially uniform, the net force is zero. Since there is no pressure differential across the spool axially and the spool has no mechanical connection to the piston, very little force is required to position the spools of all five mechanisms. For routine shutdown of the reactor, the spool may be driven to the left until the shim-safety plate is adjacent to the fuel.

When the reactor protection system operates to shut down the reactor, it reduces the current to the solenoid shutdown valve cutting off the water flow to the control mechanism and diverting it to the cooling water in the reactor pressure vessel. When the water flow to the control mechanism is cut off, the pressure differential across the piston is reduced to zero and the spring forces the shim-safety plate to the left toward the fuel (see view b of Fig. 3.4). The shutdown action is due to the spring force rather than gravity and is therefore independent of the control mechanism orientation.

To operate the reactor again, it is necessary to reestablish flow through the mechanism and to drive the spool to the left to engage the piston.

3.7. Fuel Element Mounting

The lower central fuel elements are bolted to the internal cylinder at the horizontal reactor midplane, and the upper ones are supported by the tube extending up from the control mechanisms housing. The upper ends of the 12 annular elements are mounted on the outside of the internal cylinder by means of screw-operated wedges which expand into beveled rectangular openings. The lower ends of the annular elements fit into rectangular guides which hold the elements against the outside of the cylinder but do not restrict vertical movement caused by thermal expansion. The spherical control mechanism housing and the inner surface of the fuel annulus are separated by 1/4-in. layer of water.

3.8. Outer Reflector Region and Shields

Outside the fuel annulus but separated from it by 1/2 in. of water is a spherical shell of aluminum which forces the cooling water to flow through the annular fuel elements. This shell is part of the core reflector and may be in the form of a 3/4-in.-thick aluminum shell (see Fig. 3.5) followed by water or a thin aluminum shell followed by other materials, such as lead and boral (see Fig. 3.1). One such reflector consists of 2-in.-thick lead shielding following by a 1/4-in.-thick shell of boral (aluminum-clad suspension of boron carbide in aluminum). This lead-boral region is canned in aluminum and the lead is bonded to the aluminum.

Gamma-ray shielding immediately above the central fuel elements is provided by a permanent 2-ft layer of lead shot and water contained within the inner cylinder. This shielding is penetrated by 133 helical tubes through which the cooling water enters the core. Additional shielding is provided by water which floods both the central cylinder and the reactor tank above the core.

Shield plugs (see Fig. 3.1) that can be filled with any specified material are mounted above the outer elements. During many of the experiments in which the TSR-II will be used, a specially designed shield will be mounted outside the reactor tank, in which case the material in the shield plugs will match the materials in the special shield. Two of these special shields, COOL-I and COOL-II, are shown in Fig. 3.5, along with their shield plugs. The various outer reflector regions and shields external to the reactor pressure vessel are described in Appendix A.

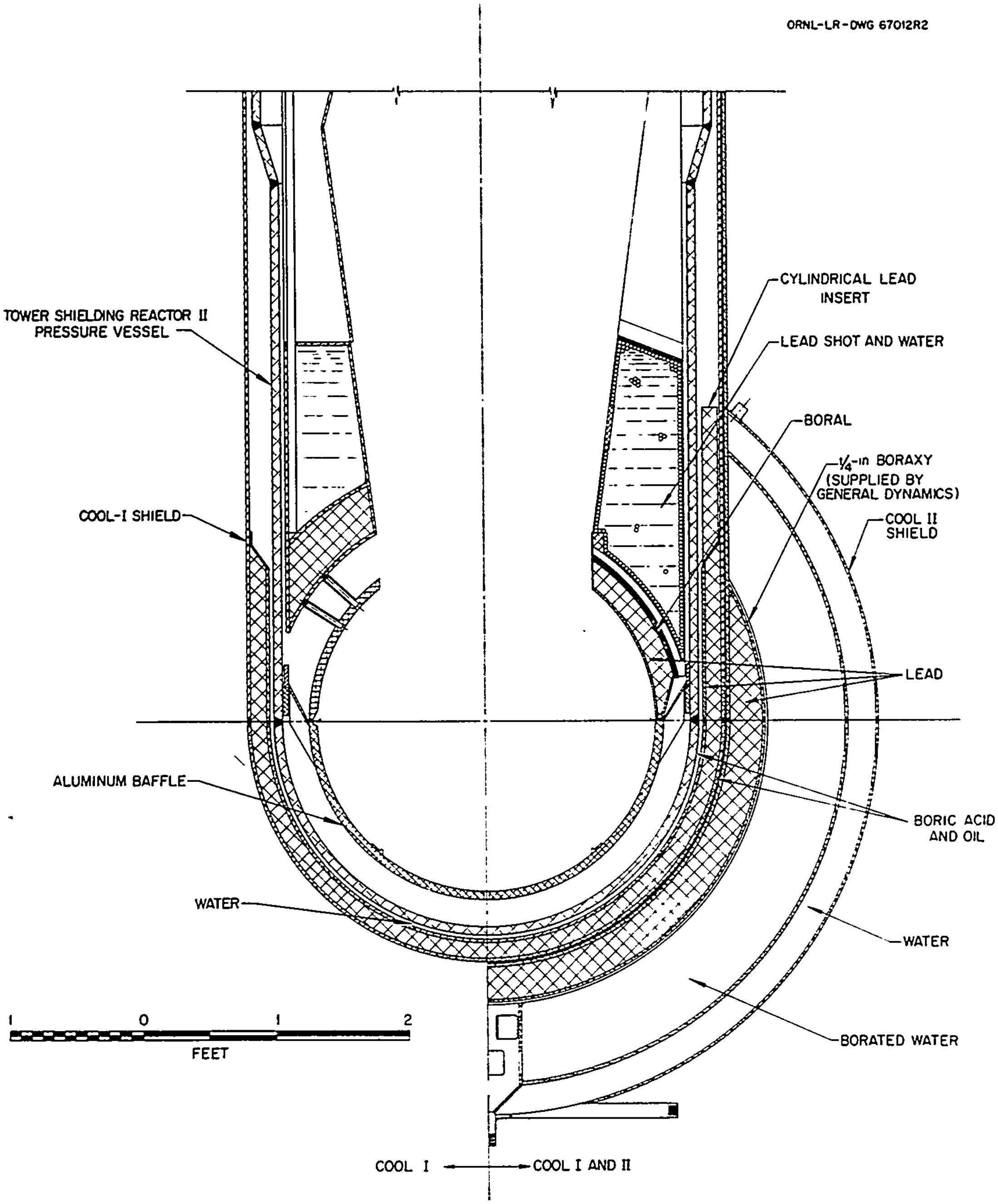

Fig. 3.5. Shield configuration.

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3. Mechanical Engineering, July 1959, American Society of Mechanical Engineers.
4. T. W. Pickel, W. A. Bush, and B. W. Wieland, Review of Stress Calculations for TSR-II, unpublished report (May 27, 1963).

4. REACTOR PHYSICS

4.1. Reactivity Considerations

The fissionable material in the TSR-II core is arranged in a spherical annulus so that the leakage radiation will be spherically symmetric, and the neutron-absorbing control plates are located in a central nonfueled region so that the perturbation of the symmetrical leakage will be as small as possible (see Chapter 3). This configuration requires a large critical mass but there is a limited excess reactivity. The reactivity change associated with various reactor parameters, the prompt neutron generation time, and the expected core life are discussed individually in the following sections.

4.1.1. Core Loading and Initial Reactivity

The loaded mass of the TSR-II is 8.370 kg of ^{235}U. This heavy loading is necessary because of the 16.75-in.-diam nonfueled region in the center of the core.

The core and reflector parameters are adjusted so that the maximum excess reactivity with any shield arrangement does not exceed 0.019 $\Delta k/k$ (see also Section 4.1.11). It is possible to operate the TSR-II with such a low excess because the research program requires that the reactor be operated intermittently and at various power levels rather than continuously at the maximum power level.

4.1.2. Shim-Safety Plate Reactivity

The shim-safety plate drive permits reactivity to be added at a rate of between 0.05×10^{-2} and 0.16×10^{-2} $\Delta k/k$ per second. The rate is about 0.06×10^{-2} $\Delta k/k$ per second at the normal critical position. Full motion of the shim-safety plates changes the reactivity 0.038 $\Delta k/k$ (see Fig. 4.1).

4.1.3. Regulating Plate Reactivity

The regulating plate has no scram capability and is operated independently of the shim-safety plates. It is fastened to a nonrotating tube which is concentric with the geared shaft that operates the five shim-safety plates and is moved slowly up or down by means of an electric motor drive. The

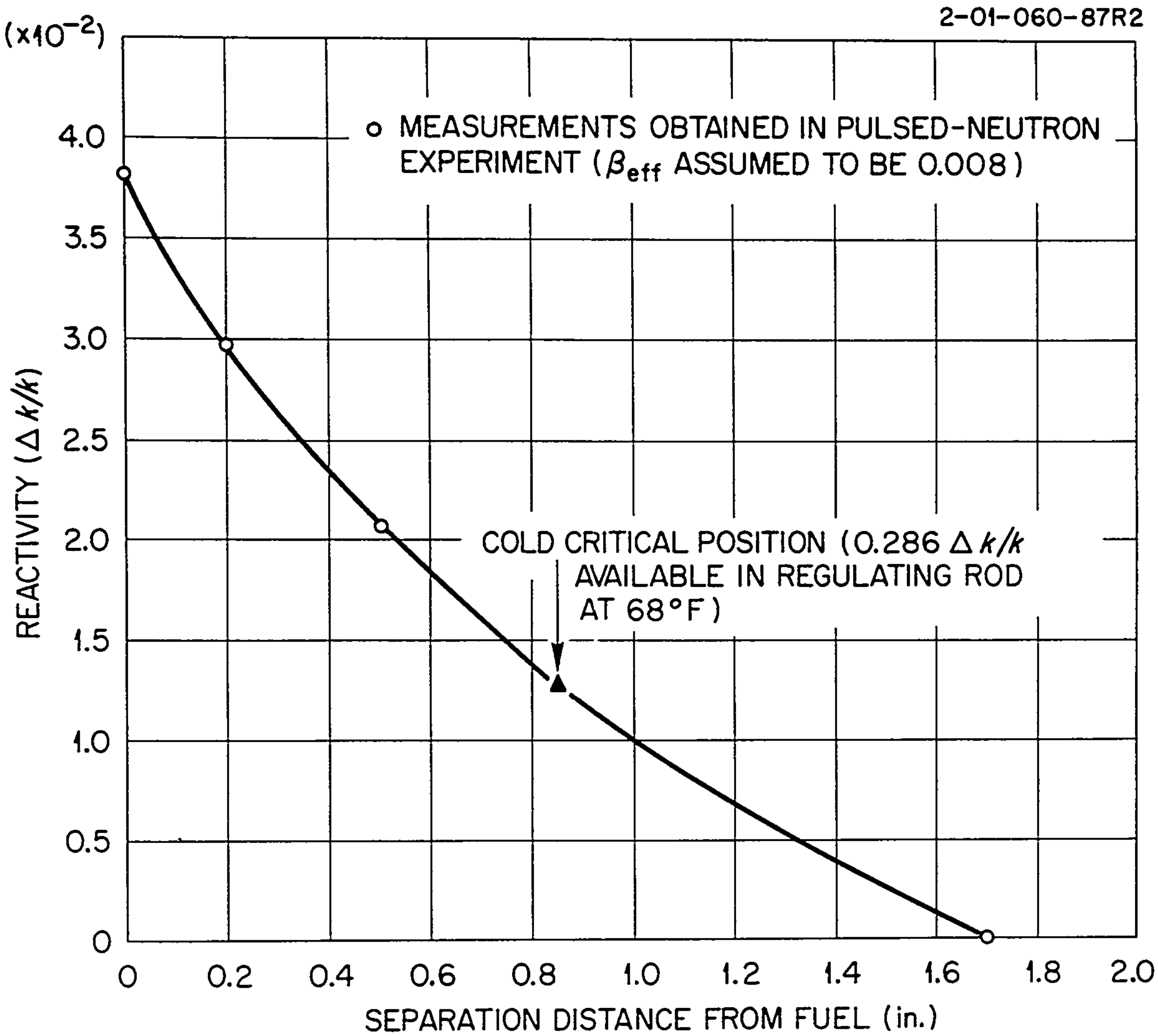

Fig. 4.1. Reactivity worth of all shim plates.

total worth of the regulating plate is approximately 0.40×10^{-2} $\Delta k/k$ and its sensitivity is between 0.034×10^{-2} and 0.108×10^{-2} $\Delta k/k$ per second; the latter value exists when the neutron absorbing plate is closest to the fuel. The rate at 1 in., the position in which the plate is normally used, is about 0.047×10^{-2} $\Delta k/k$ per second (see Fig. 4.2).

4.1.4. Temperature Coefficients

The reactivity change associated with the isothermal temperature change for the core geometry in Fig. 3.1 is shown in Fig. 4.3. The temperature coefficient ($\Delta k/k$ per ^{o}F) varies from -0.67×10^{-4} at 50^{o}F to -1.24×10^{-4} at 145^{o}F and the temperature defect ($\Delta k/k$) from 70 to 140^{o}F is 0.007.

4.1.5. Xenon Poisoning

When the reactor is operated at 1 MW, the reactivity effect of the xenon poisoning at equilibrium is -0.76×10^{-2} $\Delta k/k$. Allowing a normal weekend for decay the residual xenon effect is negligible.

Since normal operation tends to short periods of operation at various power levels an equilibrium value based on cyclic operation is more likely. Assuming for 1 MW operation that the reactor is on for 2 hr off for 2 hr for 16 hr a day five days a week, the reactivity effect of the xenon poisoning at equilibrium is 0.34×10^{-2} $\Delta k/k$. As for continuous operation, the residual xenon effect is negligible after a normal weekend shutdown.

4.1.6. Void Coefficients

The void coefficient of reactivity was calculated for the main fuel annulus and measured in and around the control mechanism housing. Void coefficients for the core were computed from changes in the multiplication constant produced by omitting the water (but not the aluminum or uranium) from thin spherical shells of the core. The results, in $\Delta k/k$ per cubic centimeter of void, are shown as a function of the radial position in the core in Fig. 4.4. The volume-averaged mean void coefficient is -2.15×10^{-6} $\Delta k/k$ per cubic centimeter of void in the core.

Void coefficients of reactivity were determined experimentally in the completely assembled reactor for voids both in the water pockets behind the control plates inside the control mechanism housing and in the water annulus

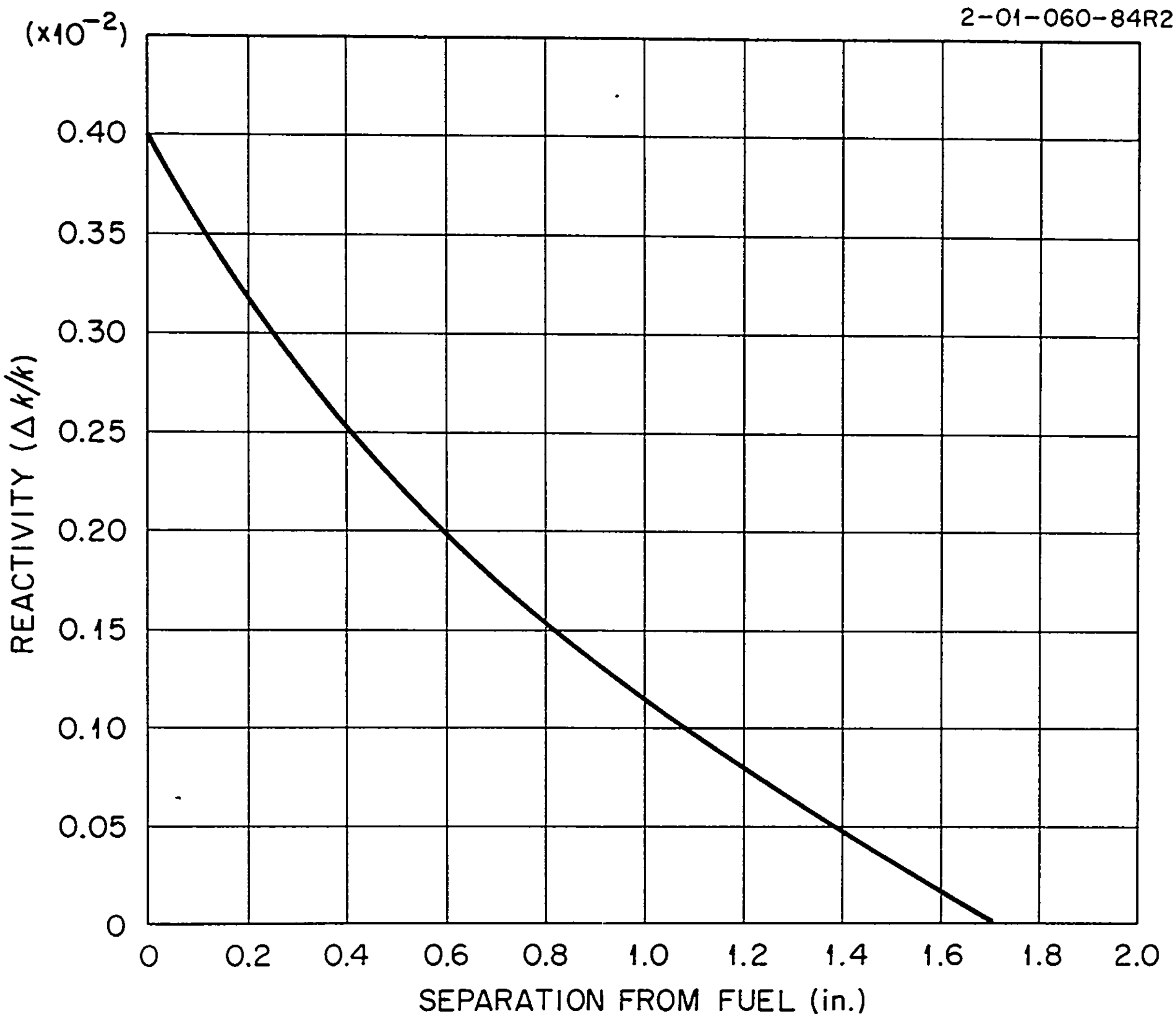

Fig. 4.2. Reactivity worth of the regulating plate.

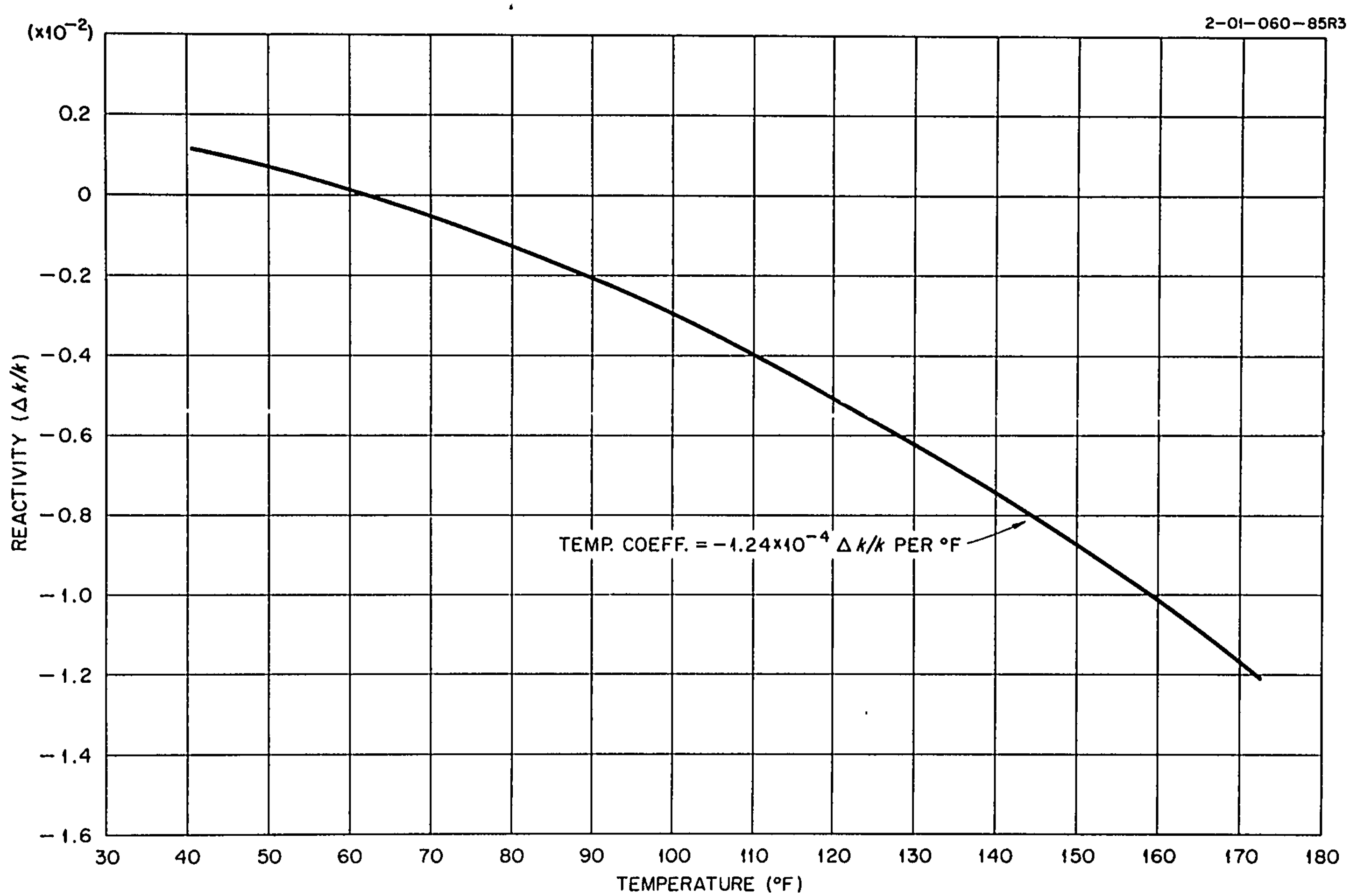

Fig. 4.3. Reactivity as a function of water temperature.

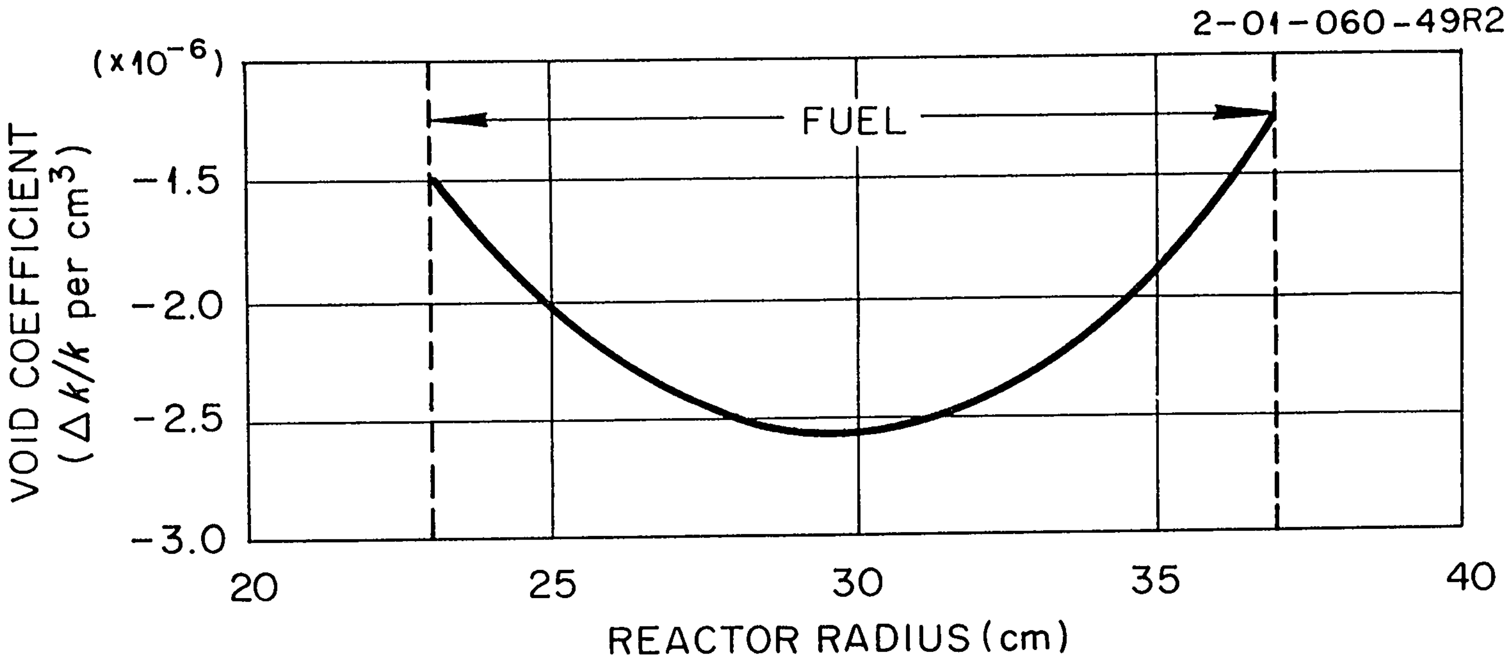

Fig. 4.4. Calculated core void coefficient as a function of radius.

between the control housing and the core region. These measurements were made by determining the reactivity change associated with displacing the water with Styrofoam. The void coefficient in the 1/4-in. water annulus outside the control mechanism housing is -6.3×10^{-7} $\Delta k/k$ per cubic centimeter of void.

Initial critical experiments performed at the Critical Experiments Facility showed that voids formed between the fuel and a poison plate in the internal reflector region are of little concern from a reactivity standpoint. Void formation in this region tends to reduce the effective water thickness between the core and the control plates and the net effect is a slight loss in reactivity. These experiments were performed to determine the worth of aluminum, control materials, and voids in the inner reflector region and led to the present design of the control mechanism housing. The geometry and the results of the experiments are shown in Fig. 4.5. The simulation of a 1360-cc void in front of a 47.6-mil-thick-cadmium dish, which had been withdrawn from the fuel, reduced the reactivity in the core by 0.036×10^{-2} $\Delta k/k$. Moving the cadmium dish to a position between the void and the fuel which simulates scramming a control plate through a void reduced the reactivity in the core an additional 0.352×10^{-2} $\Delta k/k$ (see Fig. 4.5).

Void coefficient measurements were made at both the Critical Experiments Facility and the Tower Shielding Facility for voids behind the control plates. Measurements were made behind thin cadmium, various thicknesses of boron-loaded plastic, and two thicknesses of B_4C powder in stainless steel shells. The values ranged from 0.73 to 1.06×10^{-6} $\Delta k/k$ per cubic centimeter of void, with the average value being 8.2×10^{-7} $\Delta k/k$ per cubic centimeter of void.

4.1.7. Fuel Coefficients

The reactivity of ^{235}U in the main fuel region was determined by uniformly distributing 119.3 g of ^{235}U in U-Al strips in one 30° spherical wedge of the core. The worth in the fuel-loaded lunes on the control mechanism housing was determined by replacing lunes loaded with 160 g of ^{235}U with lunes loaded with 233 g of ^{235}U. The reactivity worth in $\Delta k/k$ per kg of ^{235}U was found to be 0.027 in the main fuel region and 0.086 in the lunes.

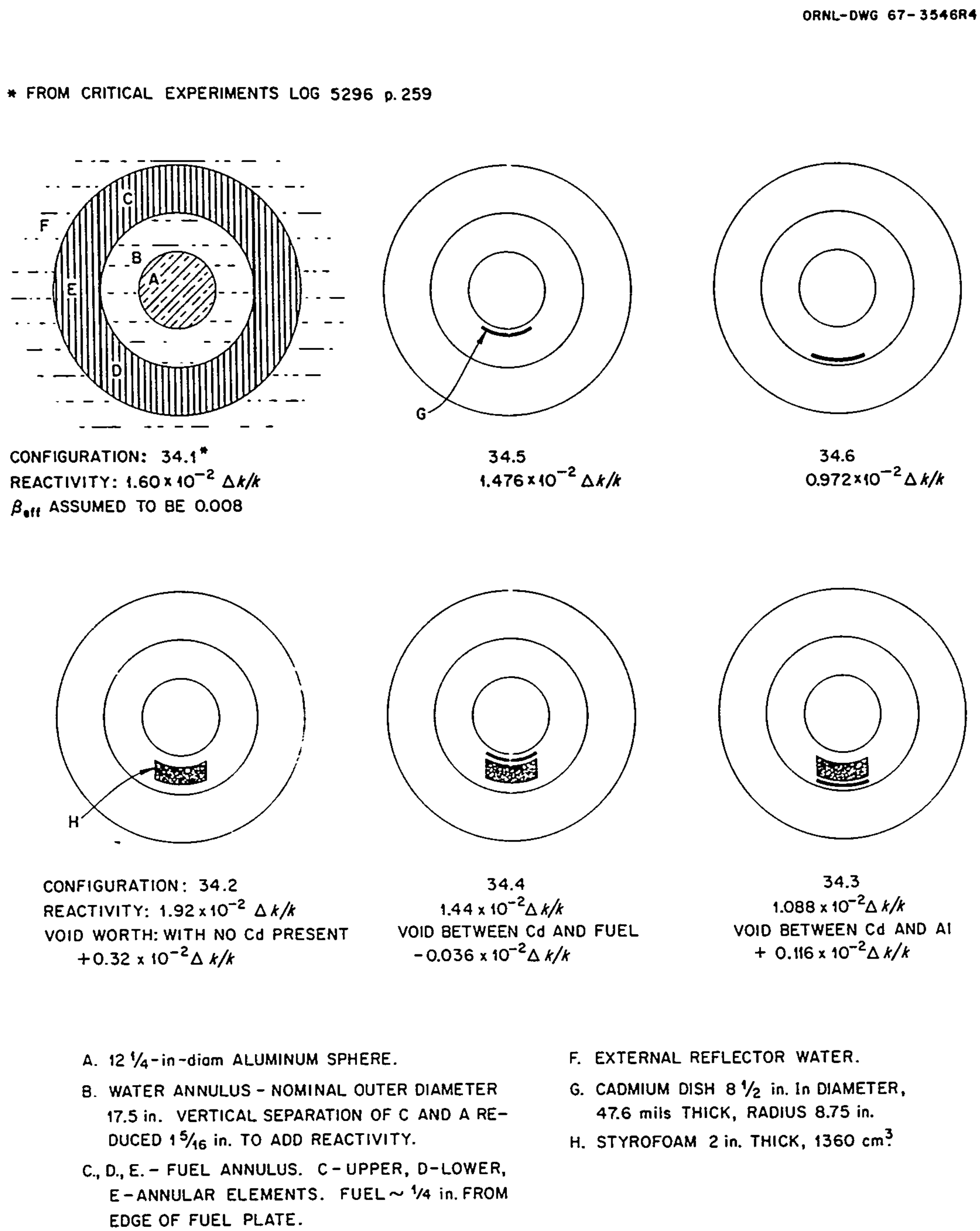

Fig. 4.5. Geometry for void measurements in inner reflector region during initial critical experiments.

4.1.8. Prompt-Neutron Generation Time

A prompt-neutron generation time of 53 μsec was obtained by using a measured ratio of the prompt-neutron generation time to the effective delayed neutron fraction, ℓ/β_{eff}, of 6.61 ± 0.16 msec and a β_{eff} of 0.0080. The ℓ/β_{eff} ratio was measured by the pulsed-neutron source technique.[1] The β_{eff} was the value determined for the BSR-I,[2] which has the same metal-to-water ratio as the TSR-II.

4.1.9. Reactivity Effect of Reflector and Shield Combinations

As described in Section 3.8, the outer reflector, which is exterior to the core but inside the pressure vessel, may be changed, or the shield exterior to the reactor pressure vessel may be changed, or they may both be changed. The basic reactor configuration is that shown in Fig. 3.1 except that the lead-boral shield is replaced with a 3/4-in.-thick aluminum shell, which has the same inner radius as the lead-boral shield, and the reactor pressure vessel is submerged in water. The reactivity changes associated with changing from the basic configuration to other combinations of reflectors and external shields are listed in Appendix B.

4.1.10. Poisoning and Fuel Depletion

The reactivity reduction due to fuel depletion and fission-product poisoning other than xenon is tabulated below for 1000, 2000, and 3000 MWhr.

	Reactivity Reduction ($\Delta k/k$)		
	At 1000 MWhr	At 2000 MWhr	At 3000 MWhr
Fuel depletion	0.00122	0.00244	0.00366
Samarium poisoning	0.00325	0.00552	0.00710
Total	0.00447	0.00796	0.01076

4.1.11. Expected Core Life

To limit the available reactivity for all cases to 0.019 $\Delta k/k$, the change due to shields different from the basic case mentioned above will be limited to +0.0016, which leaves 0.0174 $\Delta k/k$. Reserving a minimum of 0.0014 $\Delta k/k$ for control purposes, a total of 0.016 $\Delta k/k$ remains for other reactivity losses. Reducing the available reactivity by the amount of the fuel depletion and samarium poisoning leaves, for a combination of xenon poisoning and temperature defect, 0.0115, 0.0080, and 0.00524 $\Delta k/k$ for 1000, 2000, and 3000 MWhr respectively. These values indicate that the proposed operating cycle could be achieved beyond 1000 MWhr. For operation beyond 2000 MWhr some difficulty would be encountered if high-power operation is necessary when the ambient air temperature is unusually high.

4.2. Power Generation

4.2.1. Thermal-Neutron Flux Distribution

The thermal-neutron flux in the core region for the clean cold core was measured with gold foils (see curve C of Fig. 4.6). Postexperiment calculations using the multigroup GNU code[3] and IBM 704 computer were made with the control region separated the same distance from the fuel as was observed in the experiments, and the boron concentration in the control region was adjusted to make the calculated flux shape (curve A) in the core match the experimentally observed shape (curve C). Curve B shows the calculated thermal-neutron distribution when the control plates are withdrawn from 0.85 in. to 1.49 in. to compensate for reactivity reduction due to temperature rise, xenon poisoning, etc. In the latter calculation the boron concentration in the control shell was adjusted to produce the measured change in reactivity of +0.0126 $\Delta k/k$. Earlier calculations indicated that the thermal-neutron distribution in the core is affected only slightly when the external shield and internal reflector are changed. The calculated and measured values plotted in Fig. 4.6 are tabulated in Appendix C.

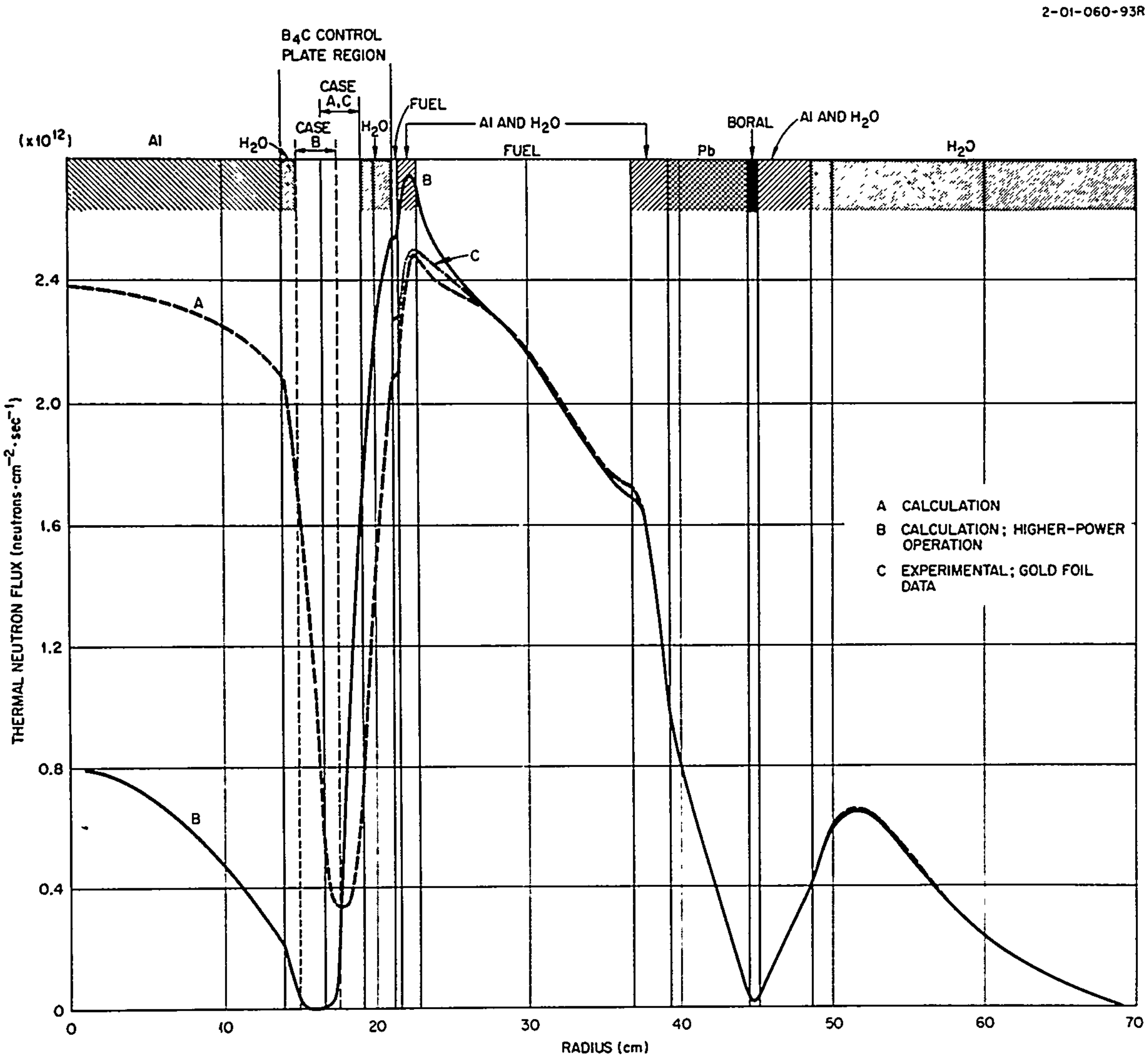

Fig. 4.6. Thermal-neutron flux in the reactor at 1 MW power level as a function of radius.

4.2.2. Power Generation in the Fuel

The radial power distribution throughout the core was calculated[4] from the measured thermal flux distribution for the clean cold condition and from the calculations with the shell withdrawn for the burnup condition of the fuel. Power generation distributions along each fuel plate were obtained with geometrical relationships for both cases. The power generation in the fuel-loaded cover plates, which was assumed to be uniform, was also calculated for both conditions.

The average power generation in the 328.7 ft^2 of the main fuel region per megawatt of reactor power was calculated to be 10,150 Btu hr^{-1} ft^{2} for clean cold operation and to be 10,123 Btu hr^{-1} ft^{-2} for high-power operation.

The ratio of the peak-to-average power generation calculated for the main fuel region was 1.20 for clean cold operation and 1.37 for high-power operation. The calculated ratio of the power generation in the fuel cover shells to the average in the main fuel region was 1.80 for clean cold operation and 2.00 for high-power operation. The ratio of the surface density of the fuel in the cover shells to that in the plates in the fuel annulus is 2.23.

References

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3. C. L. Davis, J. N. Bookston, and B. E. Smith, *GNU-II, A Multigroup One-Dimension-Diffusion Program*, GMR-101 (1957).
4. L. B. Holland and J. W. Wilson, *Calculations of the Power Distributions in the TSR-II*, ORNL-TM-1821 (June 14, 1967).

5. CORE HEAT TRANSFER

The design criterion used in the initial calculations[1] for the reactor was that there would be no boiling in the core for a power level of 5 MW and a cooling water flow rate of 1000 gpm. Initial mockup studies, however, indicated that baffle plates would be required to achieve adequate flow distribution; the baffle plates finally developed reduced the total cooling water flow rate to 800 gpm. Also, machine calculations of the afterheat generation in the fuel elements under the worst loss of coolant accident conditions indicated that the power must be limited to 1 MW for periods of 75 hr. Under these conditions the maximum temperature of the fuel, which occurs in only 0.39% of the fuel, will be 250 Fahrenheit degrees below the melting point.

5.1. Flow Distribution

The achievement of acceptable flow distribution in the coolant channels in the TSR-II was complicated by the unique geometry of the core and a low average flow rate per channel. The various methods used to observe the flow patterns and to measure the velocity of the cooling water and the final flow-rate distributions obtained are discussed below.

5.1.1. Baffle Plate Development

Since the flow rates in the various coolant channels of the reactor range from laminar to turbulent, it was first necessary to investigate the transition from laminar to turbulent flow. This was done for one channel in an annular element and it was found[2] that turbulent flow initiates smoothly (see Fig. 5.1), which means that the reactor may be operated in the transition flow regime.

Two methods were developed to measure flow rates: one for the annular elements and another for the upper and lower fuel elements.

The distribution of flow rates in the channels of the upper and in the lower elements was studied in a mockup of the central cylinder. Gross adjustment of the arrangement and shape of the baffle plate and the hole pattern in the plates was made by observing visually through plastic ends of sections of the central cylinder the distribution of dye or air bubbles leaving each flow channel. More definitive measurements were made with

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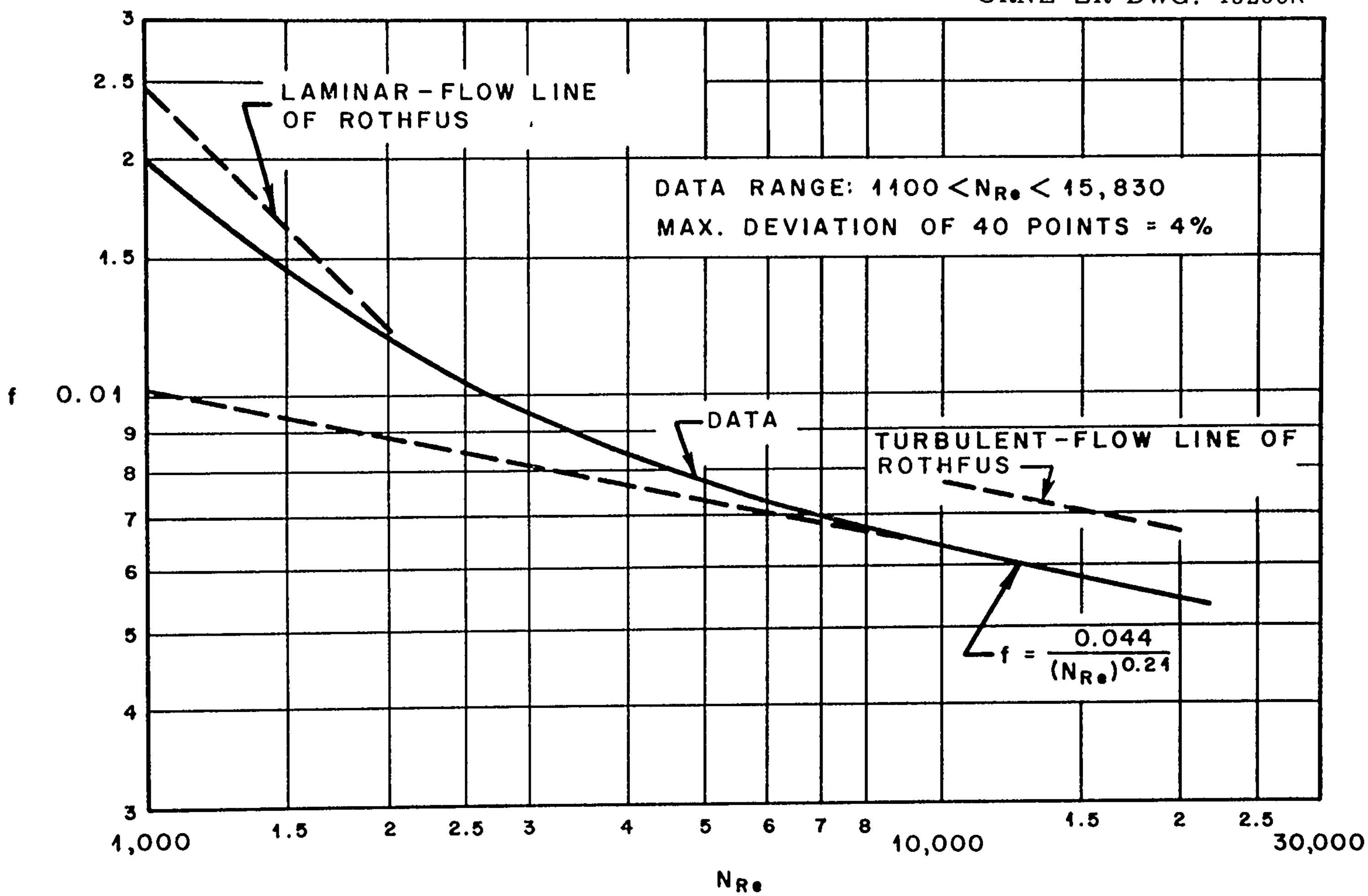

Fig. 5.1. Friction factor vs Reynolds number for the curved Parallel-plate channels (data of Rothfus from ref. 3).

conductivity probes. This method consisted in injecting a salt solution into the water flowing through the channel under test and measuring the time required for the salt solution to pass two accurately spaced probes which sensed the change in conductivity of the water. The velocity of the cooling water at some positions varied considerably. The variation at a given point was attributed to the flow not being streamline, thus causing the salt solution to be swept by eddy currents past the probes at various angles.

The cross flow or eddy currents were studied in two ways: with dye injection in the plastic annular element (described later) and with an air bubble injection in the upper and lower fuel. In channels of the upper and lower fuel elements where there was a low or varying flow rate, a few air bubbles were injected at various points in the channel, and the flow path of the bubbles was observed visually. There were eddy currents in some of the channels but there were no stagnant areas. Bubbles were quickly swept out of the channels, but since they were not always in streamline flow, variations would occur in the flow-rate indication.

Dye injection was used to measure the flow rate in channels of the annular fuel element. Dye was injected in an upstream plenum of the plastic element, and movement of the dye along the channels was photographed to obtain the flow rate. Dye was also injected at various positions along the entrance opening of individual channels so that the cross flow could be observed. Again there was evidence of cross flow but not of stagnant areas.

5.1.2. Flow Distributions

The flow-rate distribution of the cooling water in the channels of the upper fuel region was studied with complete fuel elements in a full-scale mockup of the central cylinder (see Fig. 3.1). To mock up the control mechanism housing a hemisphere of clear plastic was placed below the upper fuel elements. A tapered-thickness baffle plate (see Fig. 5.2) was placed between the elements and a mockup of the lower portion of the lead-water shield. The flow rate of the water through the mockup was 800 gpm, which is the flow rate that will be used in the reactor. All the cooling water passes through helical tubes (133) which penetrate the lead-water shield.

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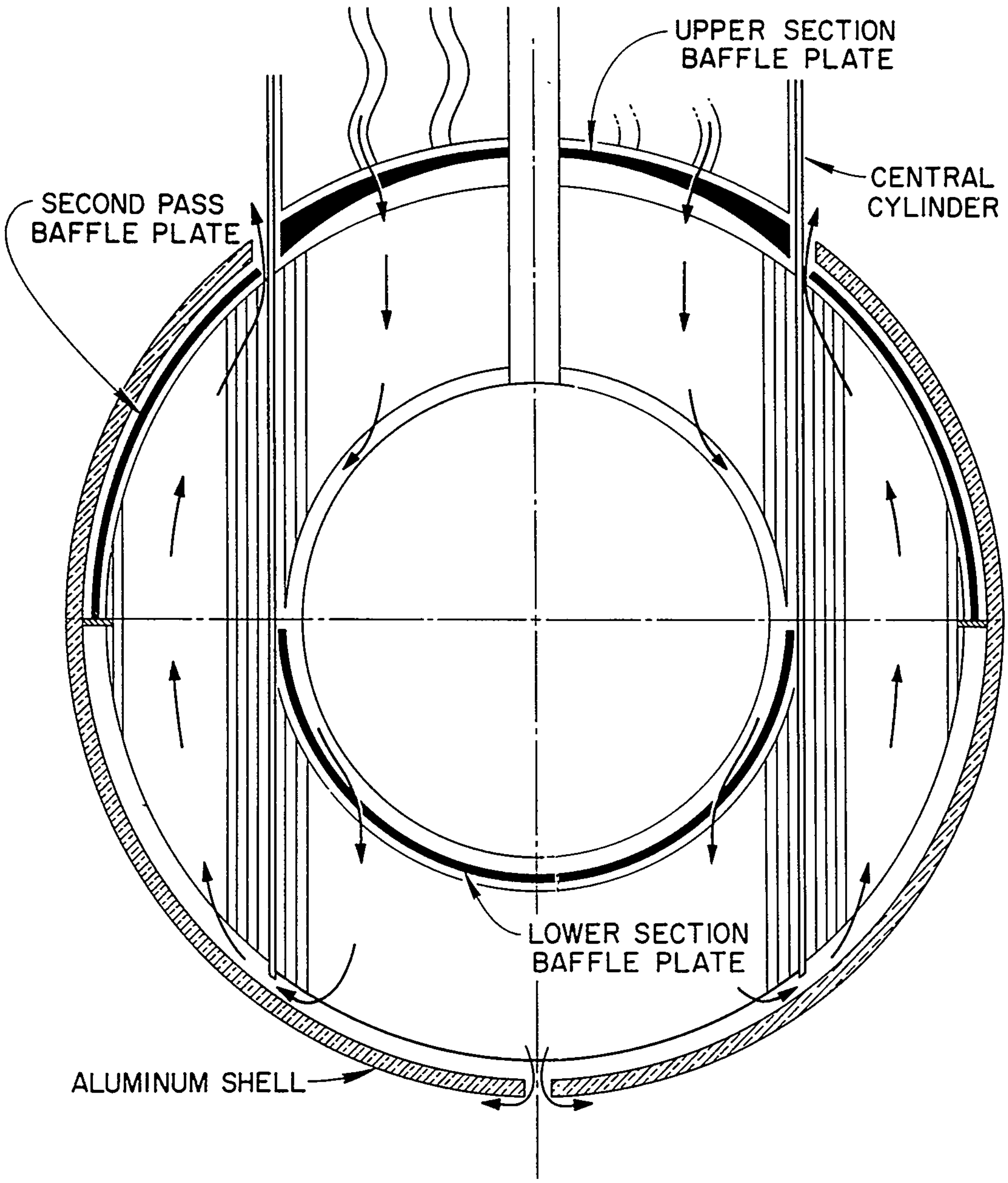

Fig. 5.2. Location of baffle plates relative to the core.

The arrangement of the exit openings of the helical tubes through the lead-water shield is not symmetrical in each quadrant, and therefore the hole patterns in the tapered-thickness baffle plates are different for each quadrant.

The flow-rate distributions obtained by the use of the tapered-thickness baffle plate are shown in Figs. 5.3 and 5.4. Figure 5.3 gives the results of two sets of measurements in one quadrant, and Fig. 5.4 shows the results of a single set of measurements in each of the other three quadrants. Each point represents the arithmetic average of 20 readings at the same location. The extremes of the readings in Fig. 5.4 are shown by error bars.

A satisfactory distribution of the cooling-water flow-rate distribution through the lower fuel elements was obtained in the same central cylinder mockup that was used for the upper elements. A clear plastic cover was used to simulate the aluminum shell below the fuel (see Fig. 3.1), and an actual control mechanism housing was mounted above the fuel. A hemispherical baffle plate of uniform thickness was placed between the fuel elements and the control mechanism housing (see Fig. 5.2). This baffle plate was easier to develop than that for the upper fuel elements because of the cylindrical symmetry in the flow paths entering and leaving the lower fuel. The flow-rate distribution of the cooling water is shown in Fig. 5.5

A plastic model of a single annular element was used for developing the exit baffle plate for each element in the annular fuel region. The model had flat rather than curved plates, and the total flow rate through the element was that which is equivalent to a total flow rate through the reactor of 1000 gpm. Because the development of the baffle plate was accomplished with a single fuel element, the flow rate in selected channels of actual elements was checked in the complete assembly of the annular fuel elements in the reactor pressure vessel. The experimentally obtained velocities for a total flow rate of 1000 gpm are shown in Fig. 5.6. Also shown are curves for a proportionate reduction per channel for total cooling-water flow rates of 800 and 400 gpm.

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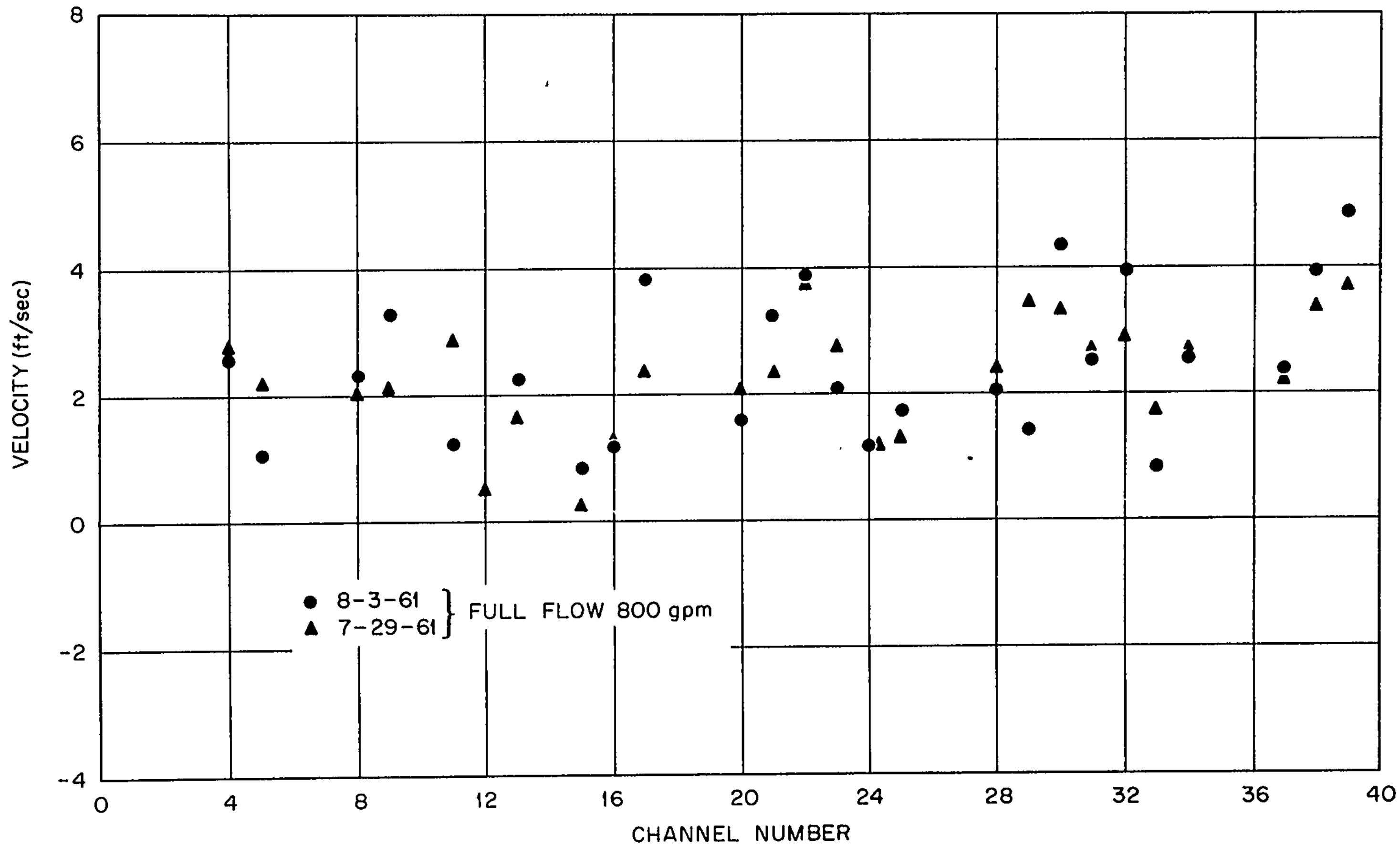

Fig. 5.3. Cooling water flow distribution in Quadrant No. 2 of the upper fuel.

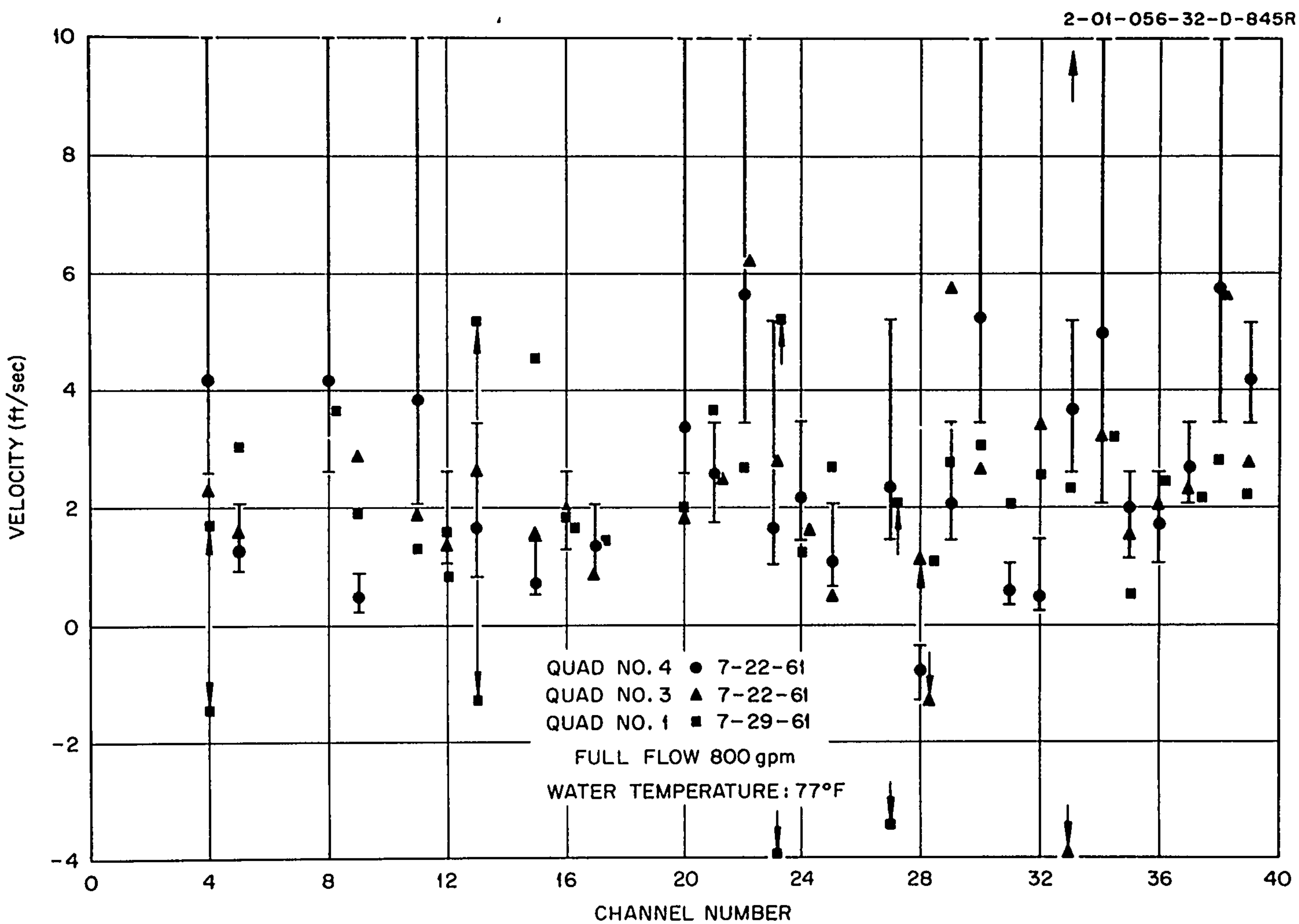

Fig. 5.4. Cooling water flow distribution in quadrants Nos. 1, 3, and 4 of the upper fuel.

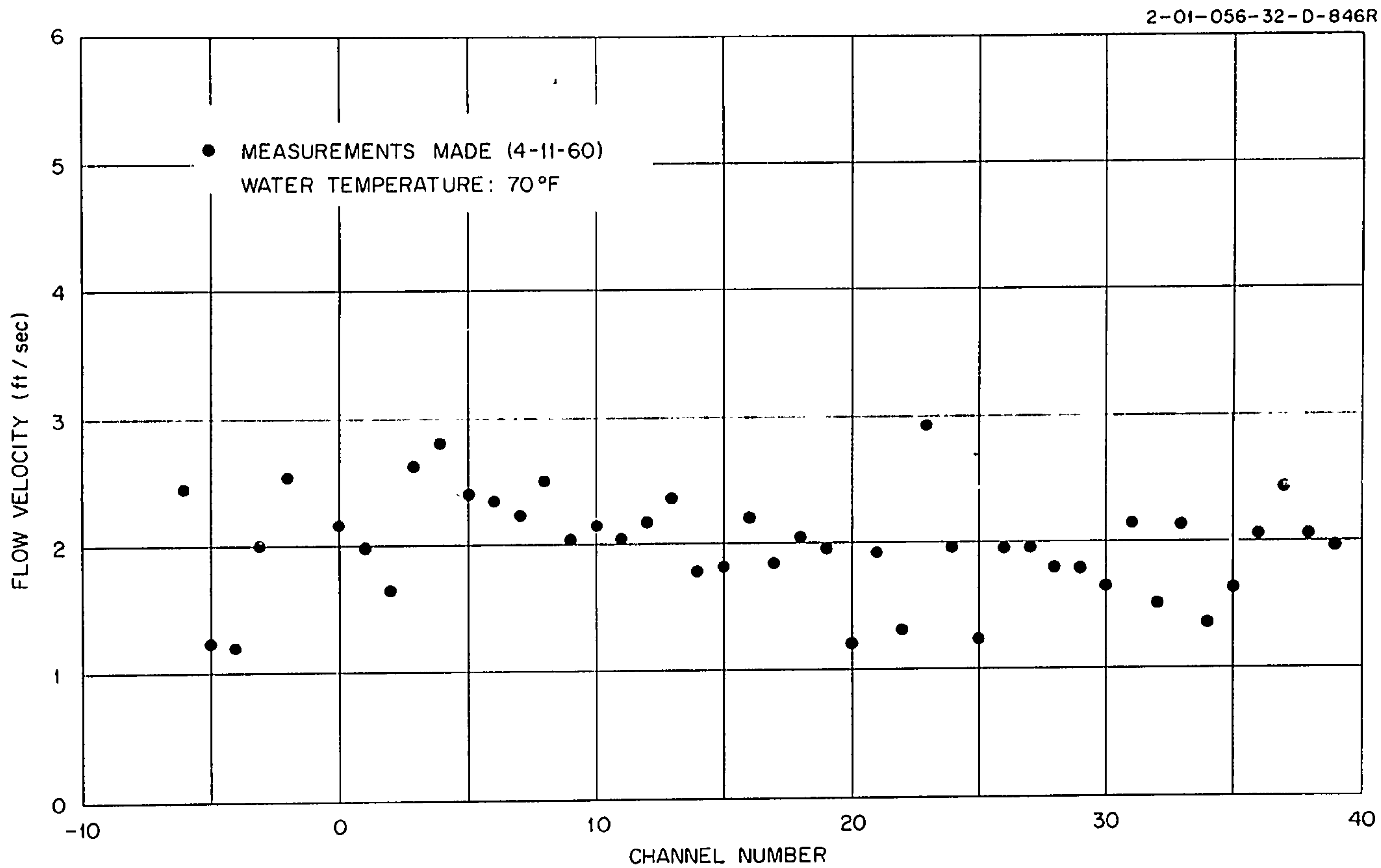

Fig. 5.5. Cooling water flow distribution in lower fuel (Total Flow 800 gpm).

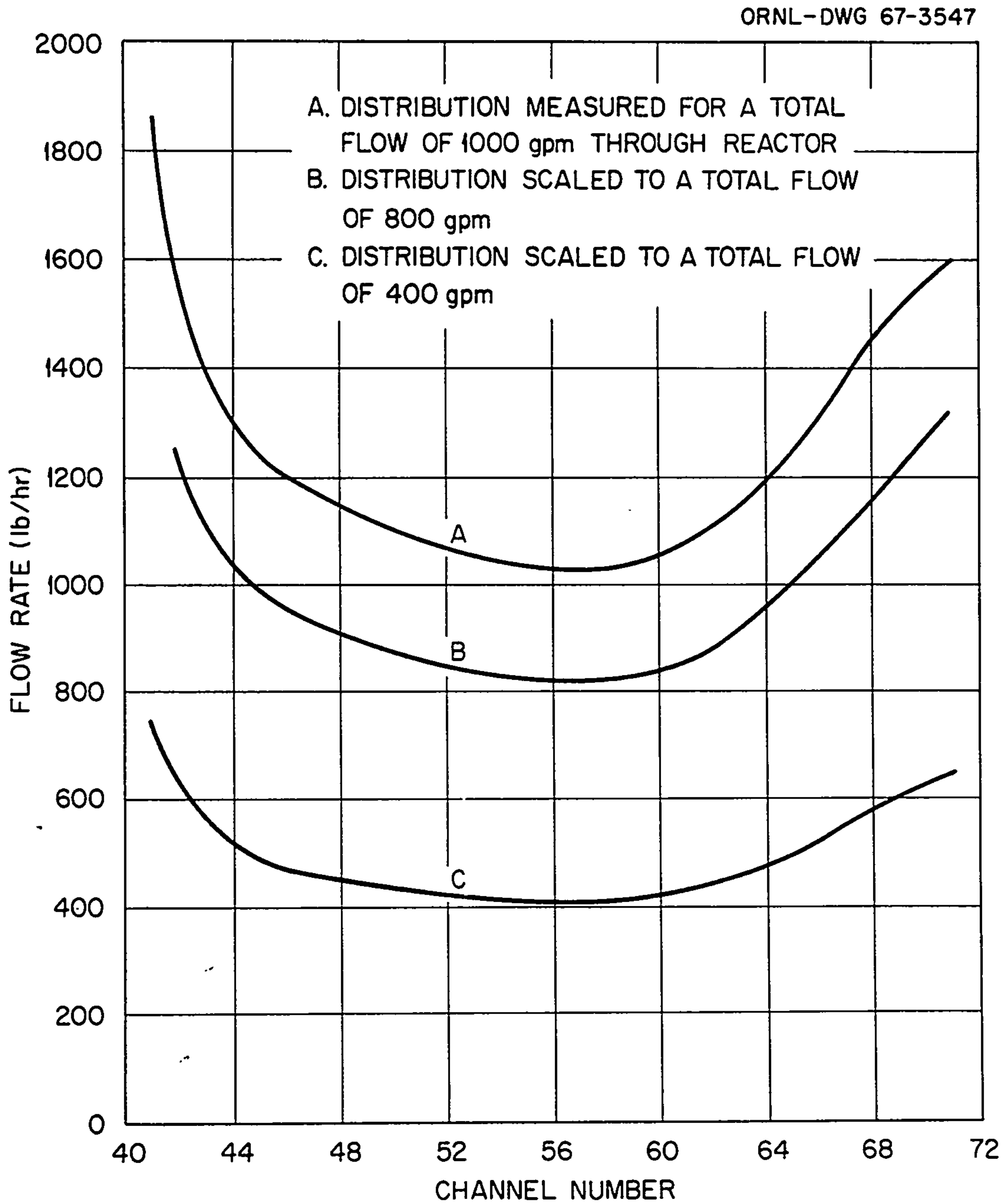

Fig. 5.6. Flow through the cooling channels of an annular fuel element.

5.2. Temperature-Distribution Calculations

The maximum temperature for the four fuel-bearing regions was calculated using the power distribution described in Section 4.2.2 and the flow distributions shown in Figs. 5.4, 5.5, and 5.6.[4] Since the minimum pressure in the core during normal operation always exceeds 36.5 psig,* the design saturation temperature was set at 274°F. The nominal operating conditions are a power level of 1 MW and a total cooling water flow rate of 800 gpm. To indicate the margin between normal maximum temperatures and the saturation temperature, some of the results reported are for a power level of 3 MW and a total cooling water flow rate of 400 and 800 gpm (see Table 5.1).

The heat-transfer considerations are similar for the upper and lower central elements in the first pass region; the cylindrical "plug" element is treated as a part of the lower fuel. The temperature was calculated at the lower end of the longest and hottest channel in the upper fuel element. For operation at 1 MW with a total cooling water flow rate of 800 gpm, the maximum temperature was only 137°F. Increasing the power level to 3 MW raised the maximum temperature to 161°F and then reducing the flow rate to 400 gpm which is one-half the normal flow rate raised it to 217°F.

To determine what effect an eddy flow would have on the maximum temperature, several combinations of flow rate, power level, and time in a channel were investigated. For one of these the total flow rate was set at 750 gpm, the power level at 3 MW, and the water was assumed to make the equivalent of five passes through the coolant channel before actually leaving the channel. With these conditions the maximum fuel plate surface temperature only reached 188°F.

The surface temperature of the fuel-loaded cover plates was calculated at the point where the fuel loading starts near the top of the cover plates.

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*The minimum pressure in the core during normal operation occurs when the reactor is elevated to its maximum height. The pressure which is the total of the static head due to the water in the hoses up to the sling bars plus the dynamic head required to return the water to the detention tank is 36.5 psig.

Table 5.1. Maximum Surface Temperature in Three Regions of the TSR-II Core for Three Combinations of Power Level and Total Cooling Water Flow Rate

	Total Cooling Water Flow Rate of 800 gpm		Cooling Water Flow Rate of 400 gpm
	1 MW	3 MW	3 MW
Upper Central Fuel Element	137	161	217
Fuel Loaded Cover on Control Mechanism Housing	144	187	219
Annular Element	156	205	258

This is also the point where the cooling water flow rate over the plates is a minimum. For normal conditions of a 1 MW power level and 800 gpm total cooling water flow rate the maximum plate surface temperature was only 144°F. The maximum increased to 187°F when the power level was increased to 3 MW and then to 219°F when the flow rate was also reduced to 400 gpm.

For the annular fuel element the temperature of the bulk water and of the fuel element surface was calculated at the midpoint of each fuel plate and for several points along the hottest channel. The maximum temperature in an annular element for a 1-MW reactor power level with a total cooling water flow rate through the reactor of 800 gpm using the most conservative correlation was 156°F. Increasing the power level to 3 MW raised the maximum plate temperature to 205°F and then reducing the water flow rate to 400 gpm increased the temperature to 258°F which is still below the water saturation temperature for the minimum core pressure which is 36.5 psig.

From the foregoing it is apparent that the reactor can be operated at a power level of 1 MW and a total cooling water flow rate of 800 gpm without the temperature of any point on the surface of the fuel elements approaching the saturation temperature.

5.3. Afterheat Removal Calculations

Studies were made[5] to determine operating cycles at various power levels which would preclude any melting of fuel plates in the event of the MCA* or at worst limit the melting to that which would limit the exposure to the general public to an acceptable level. One of these studies provided the temperature history of 256 regions in the core for 1 hr after an MCA after operation at a power level of 1 MW for 75 hr. This study indicated that the maximum temperature would be well below the melting temperature of the fuel plates. This study was extended[6] to 8-1/2 hours and the temperature history of the two hottest regions are shown in Fig. 5.7.

*Assumes loss of all coolant from the pressure vessel.

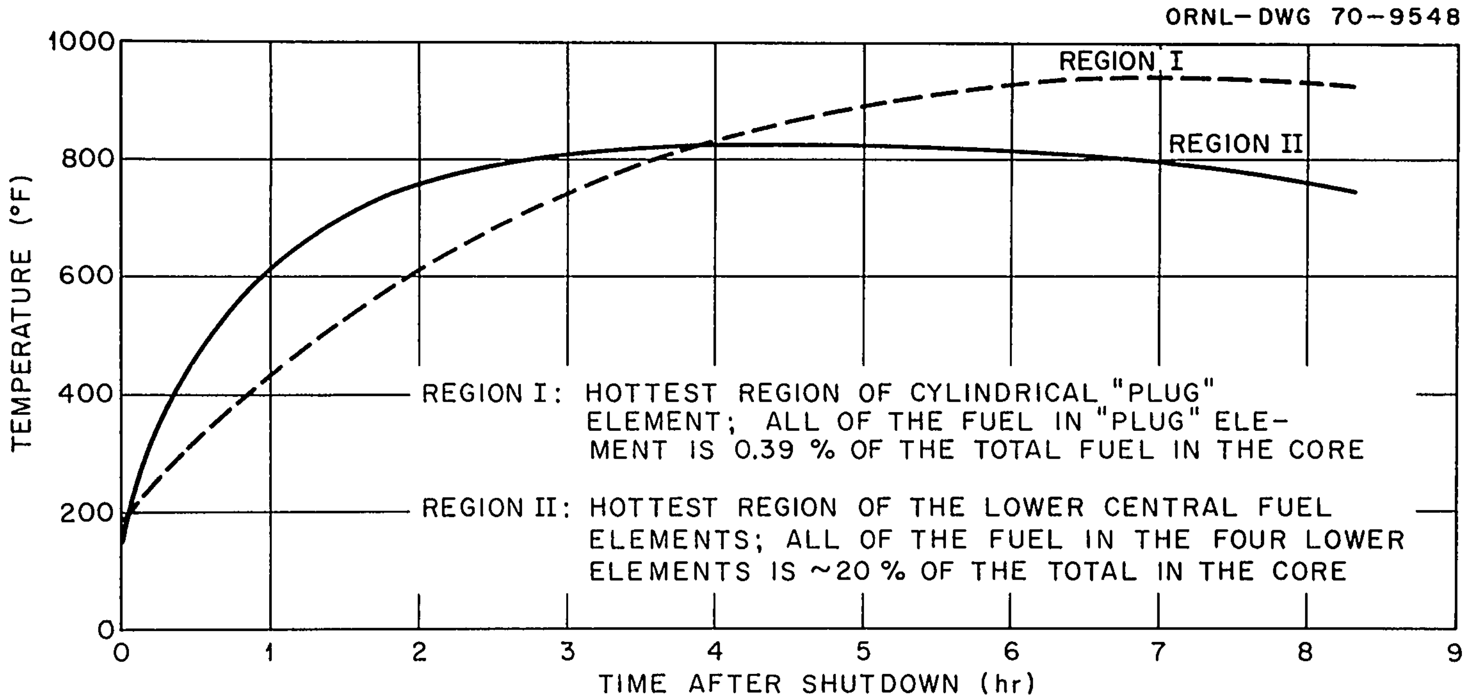

Fig. 5.7. Temperature of the hottest regions in the reactor as a function of time after shutdown for loss of coolant conditions after the reactor had been operating at 1 MW for 75-hr period prior to shutdown.

The hottest region in the core (see Region I Fig. 5.7) which reached a maximum temperature of the order of 950°F after approximately 7 hr is in the cylindrical "plug" element which contains only 0.39% of the fuel in the core. Also shown is the temperature of the hottest region of the lower fuel elements (Region II) which reached a maximum temperature of the order of 830°F after approximately 4 hr. The lower fuel elements contain approximately 20% of the fuel in the core. Since the highest temperatures in the upper and annular fuel elements are more than 200°F lower than the maximum temperature in lower elements, there is no likelihood of any core melting even for an MCA after operation at 1 MW for extended periods.

References

1. J. Lewin, TSR-II Second Pass Heat Transfer, unpublished report (January 28, 1959).
2. W. R. Gambill, TSR-II Fluid Flow Studies, unpublished report (October 21, 1959).
3. R. R. Rothfus *et al.*, AICHE Journal, 3, No. 2, 208-212 (1957).
4. J. Lewin and L. B. Holland, TSR-II Heat Transfer, ORNL-TM-1779 (July 26, 1967).
5. G. J. Kidd, Jr., A Procedure for Predicting the Effect of Loss of Coolant Accidents in the Oak Ridge National Laboratory Tower Shielding Reactor-II, ORNL-TM-2853 (May 1970).
6. Memo dated July 23, 1970, from G. J. Kidd, Jr. to L. B. Holland: "Additional Afterheat Calculations for the Tower Shielding Reactor II."

6. HEAT-REMOVAL SYSTEM

6.1. Main Flow Circuit

The heat-removal system, shown in simplified form in Fig. 6.1, consists of a 60-hp main pump for pumping demineralized water, at a rate in excess of 800 gpm, from a 1500-gal detention tank through 6-in.-diam aluminum pipe and 6-in. diam neoprene hose to the reactor and a 5-MW forced draft air cooler. The main pump operates against a 68 psi dynamic head and a fill and pressure pump operates in conjunction with a variable pressure regulating system to maintain about 5 - 10 psi in excess of the minimum necessary to keep the system full of water as the height of the reactor is varied. Pressure relief valves are located at appropriate points of the system. The water used for cooling and moderating the reactor enters the central cylinder near the top and follows the path shown by the arrows in Fig. 3.1. It flows downward through 133 helical ducts in the lead-water shield, through the upper fuel elements, around the spherical control mechanism housing, and through the lower fuel elements, and then it turns and flows upward through the annular fuel elements to the region above the core and on out of the assembly. While flowing upward the water also cools any outer reflector since some of the flow is between the outer edge of the fuel elements and the inner surface of the aluminum shell.

When the water leaves the reactor assembly, it flows to the forced-draft air cooler, where two large variable-pitch fans blow air across aluminum tube-and-fin radiators to remove the heat from the water. The pitch of the fan blades and the position of the radiator louvers can be controlled thermostatically so as to maintain a fixed temperature for the water leaving the cooler.

6.2. DC Pump for Cooling and Heating

The dc pump is a 5-hp battery-operated pump, and operates parallel to the main 60-hp pump. It runs continuously, and in the event of an electric power failure will pump water to the reactor at a rate of 40 gpm. The water from the dc circulating pump passes through two 40-kW heaters, which operate automatically with other strategically placed heaters to prevent the system

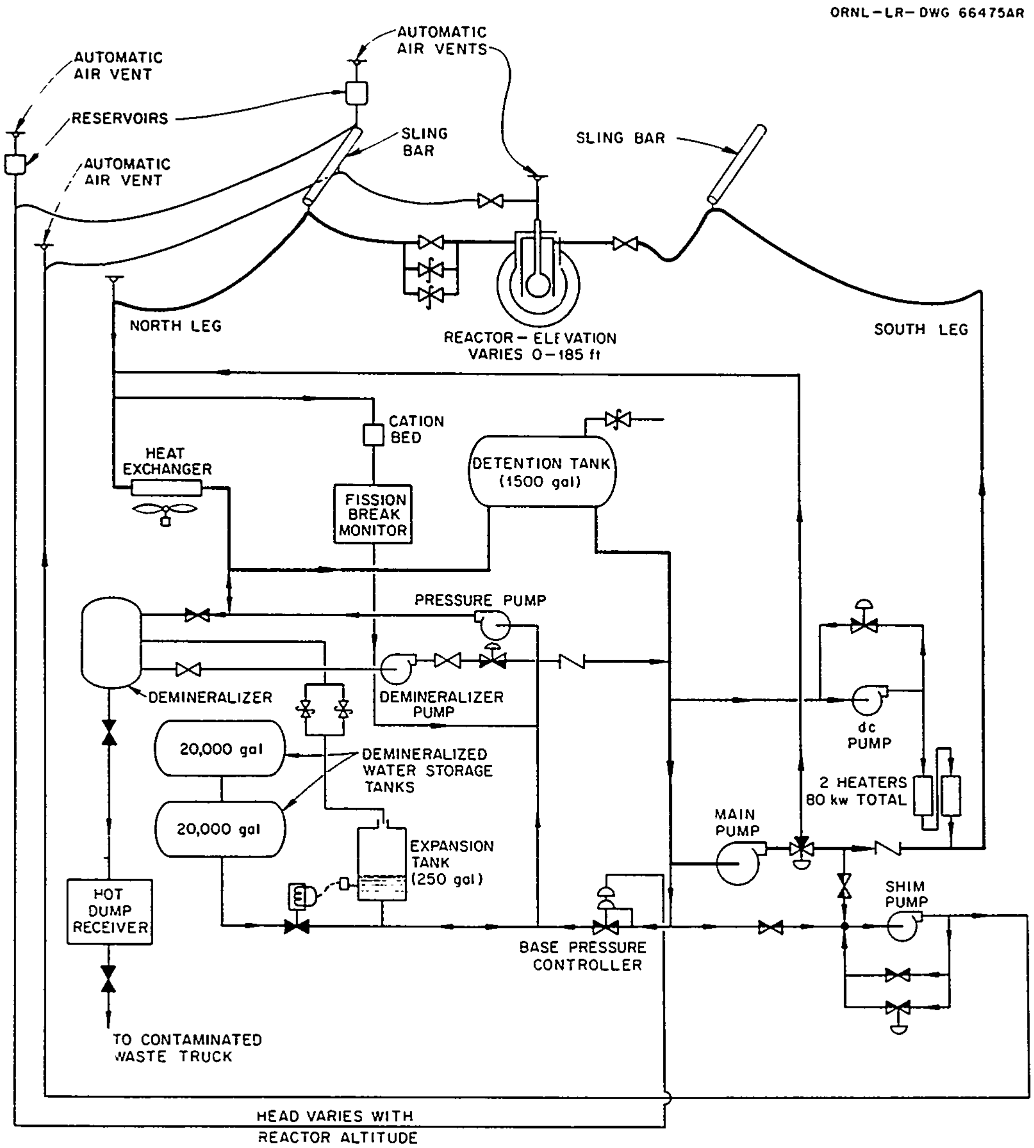

Fig. 6.1. Cooling Water System.

from freezing when the reactor is not being operated. During any prolonged loss of power in freezing weather, the water system will be drained in accordance with written procedures.

6.3. Reduced Flow Operation

Procedures require that full water flow be used during normal reactor operation; however, the system is capable of providing a reduced-flow for special operations. An air-operated valve at the main pump discharge can be controlled at the reactor console to divert all but 200 gpm of water flow from the reactor.

6.4. Shim Pump

The shim pump delivers about 20 gpm of water to the control turret of the reactor. Here the flow is divided, 5.5 gpm of it operating the reactor control mechanisms and the remainder serving to cool the control mechanism housing. Then the water joins the main coolant stream within the reactor.

6.5. Fill and Pressure Pump

The reactor is designed to be operated at elevations of up to 185 ft above ground level; therefore the water pressure at ground level must be maintained high enough to ensure adequate positive pressure at the highest points of the system at all times. To provide the minimum but adequate pressure, a pump which operates continuously takes water from a reservoir outside the main cooling system and adds it to the system, while a differential base-pressure controller (see following section) bleeds off water so that the main cooling system base pressure will be no more than the pressure on the reference side of the regulator.

6.6. Variable Pressure Regulating System

Variable base-pressure regulation is accomplished with a double-diaphragm, balanced, pressure control valve. A reference pressure is applied to the upper diaphragm of the control valve, and the cooling system base pressure is applied to the opposing diaphragm. The valve remains closed until the fill and pressure pump develops a system base pressure that begins exceed the reference pressure. At this point the overbalance in pressure will cause the valve to open and bleed off water at a rate that will maintain a balance between the reference pressure and the cooling system base pressure.

The reference pressure is obtained from a variable-height column of liquid* (see Fig. 6.2) which is contained in rigid tubing that is attached to leg I of the TSF. From this point the reference column is flexible tubing attached to a messenger cable that extends from the tower leg to the north sling bar. A small tank with an automatic air vent is installed at the top of the fixed column and at the sling bar so that whichever tank is higher will be vented to the atmosphere. The reference pressure remains constant as long as the reactor sling bar is below the top of the fixed column, and varies as a function of altitude as the sling bar is raised above this point. Annunciators alarm if the system base pressure is below that necessary to maintain the system full of water.

6.7. Makeup Water System

Ball float valves are placed at high points in the system to bleed off air or other gases so that the active loop (about 4200 gal) will be completely filled with water. Thus provision must be made to compensate for the change in volume as the water is heated or cooled. A makeup tank serves this purpose in conjunction with the pressure pump and the base-pressure controller.

A little water leakage normally occurs at the various pump seals and elsewhere. A float switch in the makeup tank operates a solenoid valve to admit more water as needed from two 20,000-gal demineralized water storage tanks. System water is never returned to these storage tanks.

6.8. Demineralizer

The demineralized water is carried by truck to the TSF. To minimize turbidity of the cooling water and activation of minerals in the water,

*Ethylene glycol (80 volume %) and water (20 volume %).

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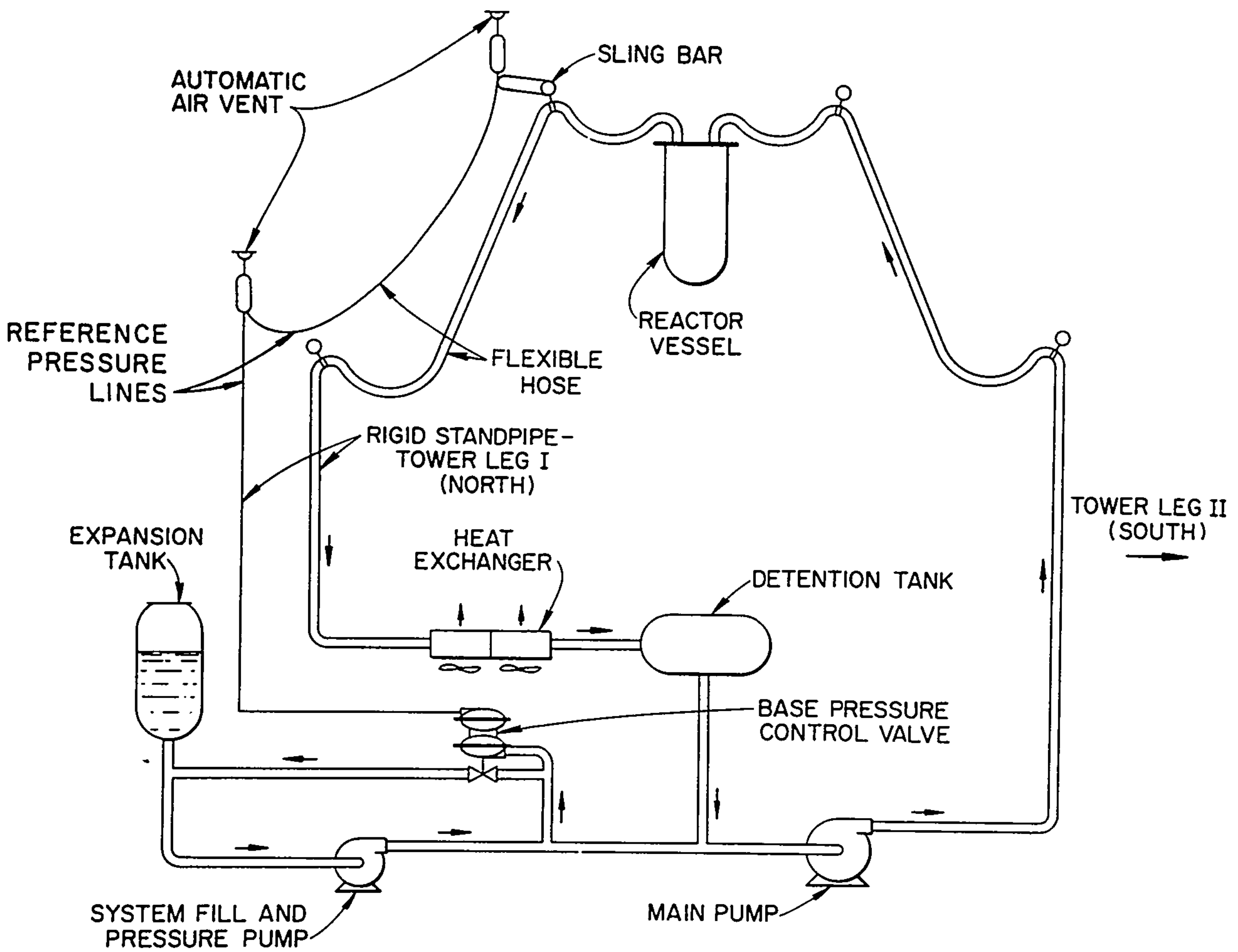

Fig. 6.2. Base pressure regulating system.

a bypass mixed-bed demineralizer is operated as necessary to maintain the resistivity of the system water above 500,000 ohm-cm. Water from the low pressure part of the reactor system passes through a throttling valve and through the demineralizer, and then is pumped back into the main loop system.

The demineralizer is regenerated as needed. The rinse residues from this process are collected in the hot-dump receiver and then transferred into a contaminated-waste truck for proper disposal according to Laboratory procedures.

6.9. Fission Break Monitor

Downstream from the reactor, at a sufficient distance to ensure that the ^{16}N activity has decayed to a negligible value, a small portion of the reactor cooling water is passed through a fission break monitor. During normal operation the gamma-ray activity of the cooling water, as indicated by the monitor, rises to within a few percent of an equilibrium value after a few minutes of operation. If fission products (primarily iodine) are present in the water, their activity will increase rapidly and trip an alarm, which is set at $\sim$1000 counts/min above the equilibrium value for full power operation.

7. INSTRUMENTATION FOR REACTOR OPERATION AND PROTECTION

7.1. General

The instrumentation systems for protection and control of the TSR-II were originally designed in accordance with the same basic philosophy that was used for the Bulk Shielding Reactor, the Tower Shielding Reactor (TSR-I), and the Pool Critical Facility.[1] The major difference between the TSR-I and the TSR-II is the inclusion of the control mechanisms in the internal reflector region of the TSR-II and the addition of a heat removal system.

In 1964 the TSR-II protection and control systems were upgraded so that, insofar as practical, there is channel isolation, dual information and action in key circuits, and both makeup and dropout actuation of the slow scram. The systems are nearly the same as those in the LITR.[2]

7.2. Logic Diagram

A block diagram indicating some of the protective and control functions is shown in Fig. 7.1. To permit the operation indicated in the circles on the diagram, all the conditions indicated in the blocks along a line from the control power to the circle must be satisfied. The conditions in the double-line blocks are met by manual operation; all other conditions are automatic. The protective and control functions to ensure orderly operation are tabulated in Table 7.1.

7.3. Neutron Flux Readout

The neutron flux values from the instrumentation for operation and/or protection that are displayed during operation are given below, together with the source from which the information is obtained:

1. The flux from source level to full power is measured by a fission chamber (constructed from ORNL design) and associated electronic equipment. The information is presented in logarithmic form and covers five decades for any given position of the fission chamber. The chamber may be moved remotely to extend its range another five decades.

2. The power level is presented on two channels: (a) a Westinghouse WL-6377 compensated ionization chamber and a log N amplifier, which provide

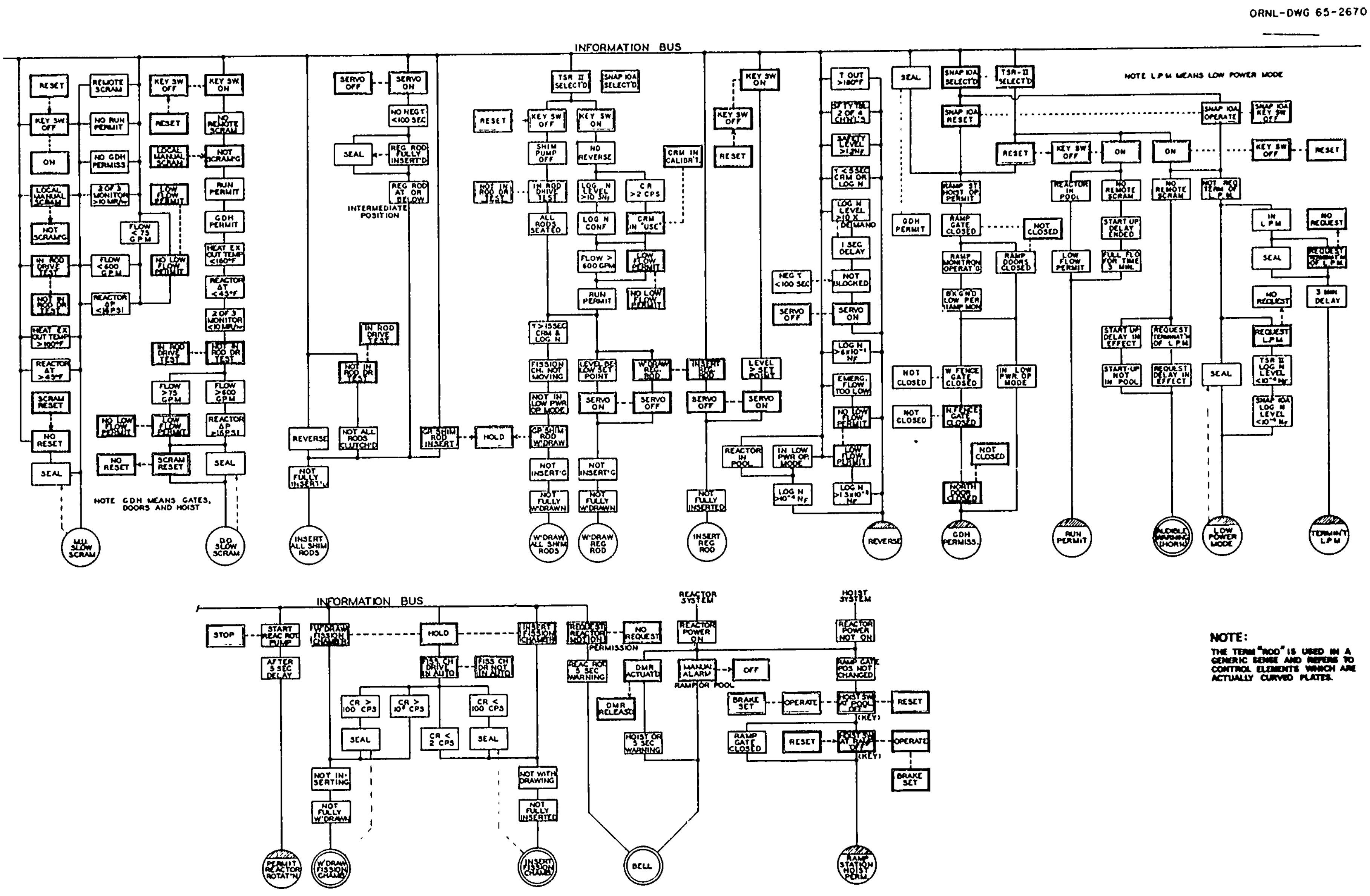

Fig. 7.1. Logic diagram of control.

Table 7.1. Protective and Control Functions

Conditions Necessary to Obtain Run Permit	Conditions Which Must Exist to Permit Shim-Safety-Plate Withdraw	Conditions Which Initiate Automatic Insertion of Shim-Safety Plates	Conditions Which Initiate a Slow Scram	Conditions Which Initiate a Fast Scram
TSF-SNAP Reactor Not Operating	TSF-SNAP Reactor Not Operating	Log N Period < 5 sec	Area Interlocks Broken	Neutron flux > 150% of Nominal Maximum
Reactor Cooling Water at Full Flow for 3 min or Low Flow Permit On	Run Permit On	Startup Channel Log Count Rate Period < 5 sec	Startup Switch Off	Period < 1 sec
Area Interlocks Made Startup Switch Reset and Warning Horn Blown for 3 min or Reactor in Handling Pool and Startup Switch Reset	Startup Channel Log Count-Rate Meter in Operate and Indicating 2 cps or Log N Power > 1×10^{-3} N_f	Reactor Outlet Water Temperature > 150°	Local or Remote Manual Scrams Actuated	
	Main Flow > 500 gpm or Low Flow Permit On	Log N Power > 1.2 N_f	Reactor $\Delta T > 14^{\circ}F$(2)	
	Log N Period > 15 sec	Two or More Safety Troubles	Shim Pump Operating for Freezeup Protection	
	Log Count-Rate Period > 15 sec	Log N Power 10 Times Servo Demand	Main Flow < 500 gpm	
	Startup Chamber Not Withdrawing (auto or manual)	No Low Flow Permit and dc Pump Flow < 10 gpm	Reactor ΔP < 10.5 psi	
	Gates, North Door Interlock Bypass Off	Log N Power Exceeds 10^{-4} N_f with Reactor in pool	Low Flow Permit On and Flow < 50 gpm Through Special Heater Circuit	
	Gates, North Door Interlock Bypass Removal Off (horn blows 3 min before it clears			

a six-decade logarithmic presentation of power level, and (b) a Westinghouse WL-6377 compensated ionization chamber and a micromicraommeter, which provide a linear power indication from the limit of detection of the chamber, less than 10^{-6} full power, up to full power.

3. Period information is obtained by electronically differentiating the output of 2(a) above, which gives d log N/dt.

4. The safety chamber currents are read from monitor instruments.

7.4. Protection System

The TSR-II primary protection system, shown in Fig. 7.2, is designed so that it shuts down the reactor whenever it receives a signal that the power level (neutron flux) is 150% of the maximum operating level or the reactor period is too short. The neutron level is sensed by three 2-in. parallel circular plate (PCP) ionization chambers, which were designed and built at ORNL. The signal is then amplified and presented to the sigma bus which responds to the highest sigma amplifier output signal. If any one sigma amplifier indicates that the neutron level is too high, it will cause all magnet amplifiers to reduce the current flowing in the coils of their respective solenoid shutdown valves, and thus produce a reactor shutdown (see Section 7.6). Similarly, a period that is shorter than 1 sec will, by the period and sigma sections of the composite amplifier, reduce all solenoid currents and cause a reactor shutdown.

7.5. Process Instrumentation

To protect the reactor and the cooling water system, the temperature and temperature change, pressure, pressure differential, and flow rate of the water are measured at various points in the system. This information is supplied to the operator and in some cases fed to automatic controls or a safety system to control the reactor operation, to operate heat-removal or heat-addition equipment, and/or to warn the operator.

Automatic control levels are set to warn the operator if the coolant temperature rise through the reactor exceeds 12°F and to scram the reactor if it exceeds 14°F or if the reactor outlet temperature exceeds 150°F. Scrams also occur if the flow through the reactor drops below 500 gpm or the pressure drop across the core drops below that equivalent to a flow rate of 500 gpm.

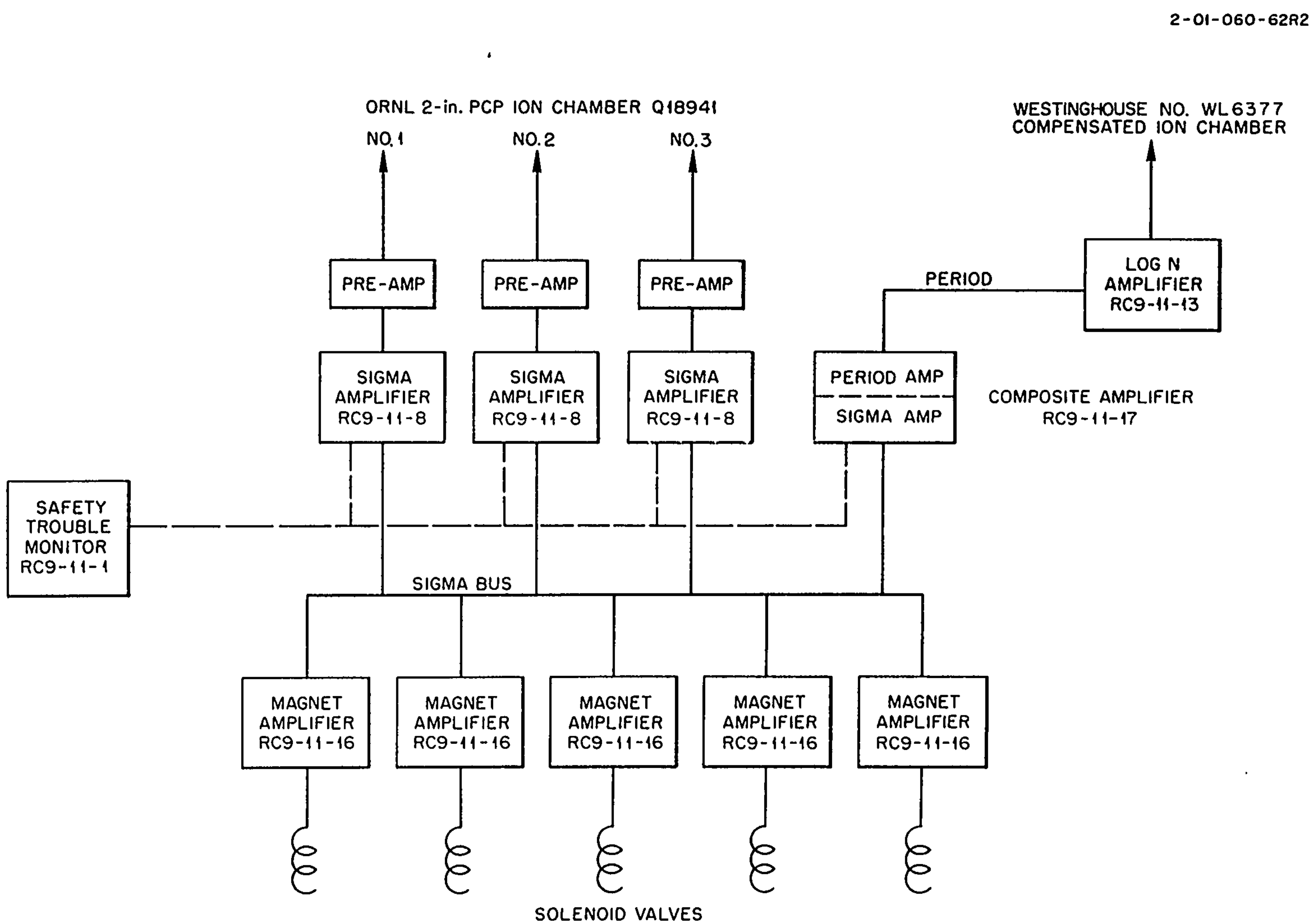

Fig. 7.2. Primary protection system.

The operator can monitor the reactor inlet and outlet temperature, the reactor temperature change (two channels), the heat exchanger outlet temperature, and the temperature at five other points in the cooling lines.

The operator can control the cooling water system temperature by operation of louvers on the heat exchanger, by operation of heat exchanger fan motors, by changing the pitch of the fan blades, or by setting automatic controls to maintain the heat exchanger outlet temperature.

Procedures exist for preventing damage to the system in cold weather. Pump operation is maintained during operating and nonoperating periods, and integrity of the system is checked regularly. Pipe insulation and automatically controlled auxiliary heaters in the system maintain the temperature of the water at a safe level.

A time-delay interlock prevents the reactor from being started before the main pump has circulated water through the reactor at full flow for a period of 3 min, the time required for the water to make one complete pass through the system; such mixing will help smooth out any temperature gradients. After startup, reduction of the main-pump flow to below approximately 500 gpm will produce a scram. Likewise, if the flow from the dc circulating pump falls below 10 gpm, a signal is produced to cause automatic insertion of the shim rods. Both minimum water-flow requirements may be bypassed for special heat-power reactor tests. For such operations an additional control command prevents the power level from rising above 75 kW.

7.6. Solenoid Shutdown Valve

Separate water lines lead from a common manifold to the five independent shim-safety-plate control mechanisms. Each line contains a throttling valve and a solenoid-shutdown valve (see Fig. 7.3) which responds to electric current from a magnet amplifier. At reactor startup the current through the solenoid coil is at its maximum, and the valve permits water (0.5 to 2.0 gpm) to flow to its associated control mechanism for operation. As the reactor power increases, the current is reduced until at 150% of the maximum permissible reactor power the current is insufficient to hold the valve open. When the solenoid valve drops out, it not only restricts the water flow to the mechanism but also diverts the flow to a bypass line.

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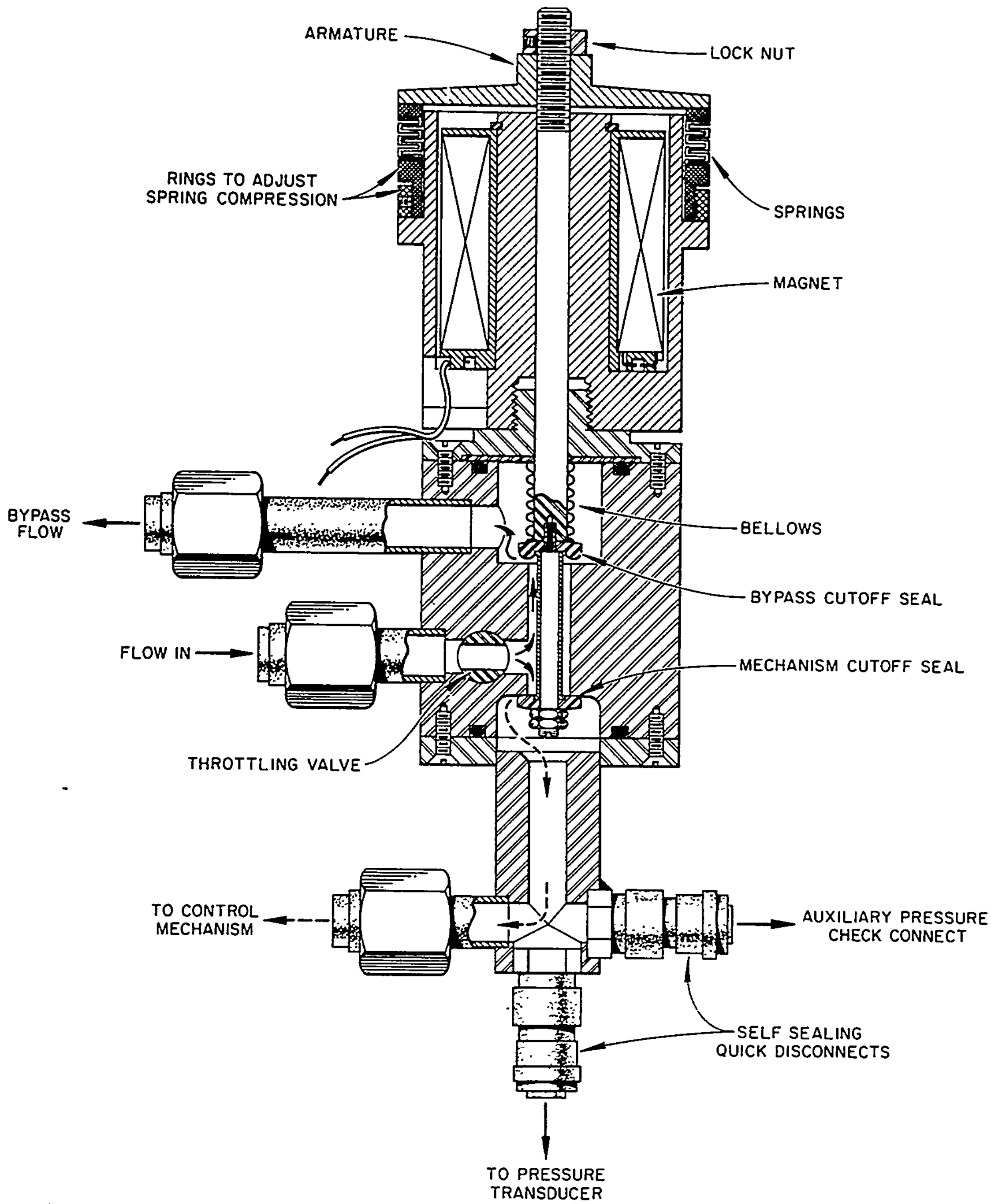

Fig. 7.3. Solenoid shutdown valve.

Instruments at the console show the solenoid current and the water flow to each mechanism.

7.7. Shim-Safety Plate Seat Information

A signal from a seat indicating system actuated by each of the five shim-safety plates indicates to the operator, by means of panel lights, when each plate is in the shutdown position adjacent to the fuel. The signal is initiated when a plate strikes a leaf spring, which opens a tiny nozzle on the control mechanism housing (see Fig. 3.3) and permits water to flow. Water from the main 6-in. reactor inlet cooling line is supplied to the nozzle through a bypass line. Turbine flowmeters monitor the flow and transmit the readings to meters on the console. Magnetic contacts on each console meter actuate a light to indicate that the plate is seated when the flow exceeds a preset value.

7.8. Shim-Safety-Plate Clutch Information

Shim-safety-rod pickup is a magnetic operation in most reactors, and engagement of the magnet and the magnet keeper on a rod is denoted as clutch made. In the TSR-II, shim-safety-plate pickup is a function of the differential hydraulic pressure in a control mechanism, and clutch information is indicated individually for the five plates by metering the flow to each control mechanism. Flow is monitored with turbine flowmeters which drive indicators at the console. When a plate is properly engaged, its flow remains within preset limits, which is indicated by its clutch light and meter on the console.

7.9. Plate Position Indicators

Upper- and lower-limit switches actuated directly by the plate drive indicate the fully inserted or fully withdrawn plate positions. A dual selsyn system is used to indicate the position of the shim-safety-plate drive, while a single selsyn system indicates the regulating plate position.

7.10. Servo System

A servo system is provided to maintain the power level at any point within the operating range. The servo is of the on-off type.

7.11. Instrument Location

A pulse preamplifier in the fission chamber circuit and the safety chamber preamplifiers are located at the reactor, and the log N preamplifier is located in a sling box approximately 30 ft from the reactor. All other electronic components are located in underground Building 7702.

7.12. Neutron Source for Reactor Startup

The TSR-II core contains two fixed sources of neutrons: an Sb-Be source centered in the cylindrical plug fuel element, which is part of the lower fuel element assembly and is fixed in the reactor, and an Am-Be source, which is located in the control mechanism housing. Only the Am-Be source (emission rate of $\sim 6.6 \times 10^6$ neutron/sec) will be used in subsequent core loadings.

7.13. Personnel Access Control

All personnel entering the TSF area are issued a tally badge. Prior to operation the individual operating the reactor, in accordance with procedures in the TSF Manual, checks that all personnel are accounted for and informed of the restrictions in force during the operation. During in-air operation, personnel must remain in the underground control building except as noted below. Before operation can be initiated, a warning horn will blow for 3-minutes to alert any personnel who may have been missed in the personnel check that operation is imminent. To ensure that personnel cannot inadvertently approach the reactor area during operation, interlocks will cause a reactor shutdown if anyone opens a door to leave the control building or opens a gate in the 600-ft radius fence to enter the area.

Specified personnel may enter or leave the TSF area while the reactor is operating in air, but only under the supervision of a TSF Supervisor utilizing the instrumentation installed for that purpose when conditions approved by the RORC exist.

Special procedures in the TSF manual are used for personnel control during critical experiments and maintenance.

References

1. A. E. G. Bates, "Description of ORNL Pool Type Critical Facility," ORNL-CF-57-7-51 (July 16, 1957).
2. A. E. G. Bates, "Description of the LITR Reactor Control and Instrumentation Systems," ORNL-TM-1190 (July 16, 1965).

8. WASTE DISPOSAL

A small amount of contamination can always be expected in the cooling water that passes through the reactor during normal operation, but it will be concentrated in the demineralizer. Fission-product leakage from the fuel elements is, of course, a possible source of contamination. The reactor cooling water system is designed so that the regenerants and contamination from the demineralizer are drained into a glass-lined steel tank, which is located below and to the west of the reactor cooling water pump room. If the reactor system has to be drained, the water is also drained into this tank, which has a capacity of nearly twice that of the reactor cooling system. Another drain that is connected to the glass-lined tank is located near the reactor-handling pool so that the water which must be drained from the hoses or reactor pressure vessel will also be drained into the tank. The operation of the system is intended to prevent any chemicals or contamination from being released to run off with the natural drainage.

Whenever waste water is drained into the tank, the radiation level of the liquid is checked and the waste liquids are transported to the main ORNL waste disposal area in tank trucks according to established Laboratory limits and procedures.

APPENDIX A. DESCRIPTION AND ABBREVIATED NOMENCLATURE FOR OUTER SHIELDS AND REFLECTORS

As described in Section 3.8, the reflector materials inside the pressure vessel and the shield configurations external to the pressure vessel may be changed. The shield configuration external to the reactor pressure vessel is denoted by a name or abbreviation, followed by a letter that denotes modifications in the shield, as shown below:

Bare-A	Only air external to the reactor pressure vessel
Bare-B	Reactor pressure vessel submerged in water
CI-A	COOL-1 shield (ORNL General Engineering Dwg. M-20858-EJ-126-D) as originally fabricated (Al, 1/4 in.; Pb, 1 1/2 in.; Al, 1/4 in.) with 3/8-in. water gap between shield and pressure vessel and with the reactor suspended in air
CI-B	Same arrangement as CI-A but with a 1 1/2-in.-thick, ~12-in.-high cylindrical shell of lead added above the original fixed lead and a B_4C and oil mixture placed between the shield and the pressure vessel
CI-C	Same as CI-B but with the B_4C and oil replaced with water
CI-D	Same as CI-C but with shield submerged in water
CII-A	Outer COOL shield (General Engineering Dwg. M-20858-EJ-127-D) as originally designed (Al, 1/4 in.; Pb, 3 in.; Al, 1/4 in.; Boroxy, 1/4 in.; water, 7 in.; Al, 1/4 in.; water, 2 in.; Al, 1/4 in.) over CI-B (General Engineering Dwg. M-20858-EJ-123-D) with B_4C and oil in the 1/2-in. gap between the

COOL shields, borated water in the 7-in. inner region, and plain water in the outer region; shield can be in air or submerged in water

CII-B	Same as CII-A but with borated water in both inner and outer regions
CII-C	Same as CII-A but with plain water in both inner and outer regions
Beam-A	Beam shield (General Engineering Dwg. M-20858-EJ-106-D), steel, 1/4 in.; Pb-H_2O each 50% by vol, 16 5/8 in.; steel, 1/4 in.; water, 31 in.; steel, 1/4 in.; plain water in the inner and outer region and the 1/2-in. gap between the pressure vessel and the shield; shield can be in air or submerged in water
Beam-B	Same as Beam A except with all water drained from the shield
PW-A	Special purpose lithium hydride, depleted uranium shield designed by Pratt and Whitney

The reflector is made up of five pieces, with the major one consisting of a hemispherical shell mounted below the core. The remaining four pieces mate with each other and the lower half to form a spherical shell that surrounds the core except for a penetration by a permanent lead-water shield that is mounted directly above the core. Shaped plugs of shielding material may be mounted on the top side of each of the four outer-reflector pieces.

To identify the combination of shield configuration, reflector, and shaped plugs, the number which identifies the outer reflector and shield plug configuration is affixed to the abbreviated nomenclature for the shield configuration. For example, CI-A3 represents the original COOL-I shield with a reflector that has a 3/4-in. aluminum shell followed by 3 1/2 in. of water for the lower hemisphere, a 3/4-in. aluminum shell followed by 3 1/2 in. of H_2O, then 4 in. of Pb, 1/4 in. of aluminum, and finally H_2O for the upper hemisphere and no shaped shield plugs.

The various combinations of materials which have been used for the reflector and the shaped plugs are described in Table A.1.

TABLE A.1. Materials Used in Various Reflector and Shaped Plug Configurations

Configuration No.	Lower Reflector	Upper Reflector	Shaped Plugs
1	Al, 1/4 in.; Pb, 2 in.; boral, 1/4 in.; Al, 1/8 in.; H_2O, 1/2 in. (M-20858-EJ-132-D*)	Al, 1/4 in.; Pb, 2 in.; boral, 1/4 in.; Al, 1/8 in.; H_2O, 1/2 in. (M-20858-EJ-139-D and 142-D*)	H_2O and Pb each 50% by vol (M-20858-EJ-130-D*)
2	Same as No. 1	Same as No. 1	H_2O
3	Al, 3/4 in.; H_2O, 3 1/2 in.	Al, 3/4 in.; H_2O, 3 1/2 in.; Pb, 4 in.; Al, 1/4 in.; H_2O (M-20858-EJ-125-D*)	H_2O
4	Same as No. 3	Same as No. 1	H_2O and Pb each 50% by vol
5	Same as No. 3	Same as No. 1	H_2O
6	H_2O	H_2O	H_2O

*ORNL General Engineering drawings.

APPENDIX B. REACTIVITY WORTH OF REFLECTOR AND SHIELD COMBINATIONS

The reactivity worth of usable arrangements of existing shields and reflectors has been determined. In some cases these determinations have been made with the reactor and shield suspended first in air and then in water. In other cases determinations have been made with the water drained from the shield in order to predict the expected reactivity change associated with unplanned variations. The reactivity changes associated with all arrangements thus far determined have been smaller than predicted, and all are within safe handling limits. The change in reactivity that occurs when the reactor reflector or shield arrangement is changed from the reference configuration is tabulated below according to the abbreviated nomenclature (see Appendix A):

Shield and Reflector Configuration	Reactivity Change, % $\frac{\Delta k}{k}$ vs Bare-B3
Bare-B1	-0.21
Bare-A1	-0.29
Bare-A3	-0.31
Bare-B6 (for information only)	+0.17
Bare-B3 (reference)	0.00
Beam-A1	-0.04
Beam-A3	-0.08
Beam-B1	+0.16
Beam-B3	-0.05
PW-A1	+0.07
CI-C3	-0.01
CI-D3	+0.09
CII-A3	+0.05
CII-A4	+0.14

APPENDIX C. CALCULATED AND MEASURED THERMAL-NEUTRON FLUXES IN TSR-II

Table C.1. Calculated Thermal-Neutron Fluxes for Clean Cold Operation.
Separation between ^{235}U-loaded cover plates and B_4C control plates = 2.16 cm.

Radius (cm)	Thermal-Neutron Flux (neutrons·cm^{-2}·sec^{-1}·MW^{-1})	Radius (cm)	Thermal-Neutron Flux (neutrons·cm^{-2}·sec^{-1}·MW^{-1})	Radius (cm)	Thermal-Neutron Flux (neutrons·cm^{-2}·sec^{-1}·MW^{-1})
	x 10^{12}		x 10^{12}		x 10^{12}
1.00	2.38	22.22	2.42	45.96	0.138
2.00	2.38	22.50	2.48	46.83	0.224
2.99	2.36	22.86*	2.48	47.70	0.326
3.99	2.36	23.86	2.42	48.58	0.406
4.99	2.34	24.86	2.38	49.09	0.498
5.99	2.34	25.85	2.36	49.60	0.568
6.99	2.32	26.85	2.34	50.10	0.618
7.98	2.30	27.85	2.30	50.60	0.650
8.98	2.28	28.85	2.26	51.12	0.666
9.98	2.26	29.85	2.20	51.63	0.670
10.98	2.24	30.84	2.12	52.14	0.662
11.98	2.20	31.84	2.06	52.64	0.648
12.97	2.16	32.84	1.97	53.15	0.626
13.97	2.10	33.84	1.89	53.66	0.600
14.62	1.94	34.84	1.81	54.17	0.572
15.27	1.61	35.83	1.76	54.68	0.540
15.92	1.14	36.83*	1.73	55.18	0.508
16.57	0.552	37.46	1.68	55.69	0.476
17.20	0.366	38.10	1.52	56.71	0.414
17.84	0.340	38.64	1.28	57.72	0.354
18.47	0.446	39.37	0.968	58.54	0.302
19.11	0.726	40.22	0.792	59.76	0.254
19.65	1.21	41.06	0.626	60.77	0.212
20.19	1.60	41.91	0.468	61.79	0.174
20.73	1.90	42.76	0.318	62.80	0.142
21.27	2.10	43.60	0.176	63.82	0.114
21.35	2.10	44.45	0.038	64.84	0.090
21.43	2.10	44.61	0.032	65.85	0.068
21.51	2.11	44.77	0.028	66.87	0.048
21.59	2.11	44.93	0.028	67.88	0.030
21.90	2.30	45.08	0.030	68.90	0.012

*Boundaries of main fuel annulus.

Table C.2. Calculated Thermal-Neutron Fluxes for Maximum Reactivity Reduction Due to Poisoning, etc. Separation between ^{235}U-loaded cover plate and B_4C control plates = 3.81 cm.

Radius (cm)	Thermal-Neutron Flux (neutrons·cm^{-2}·sec^{-1}·MW^{-1})	Radius (cm)	Thermal-Neutron Flux (neutrons·cm^{-2}·sec^{-1}·MW^{-1})	Radius (cm)	Thermal-Neutron Flux (neutrons·cm^{-2}·sec^{-1}·MW^{-1})
	x 10^{12}		x 10^{12}		x 10^{12}
1.00	0.792	22.22	2.74	45.96	0.136
2.00	0.782	22.50	2.76	46.83	0.232
2.99	0.766	22.86*	2.70	47.70	0.320
3.99	0.742	23.86	2.54	48.58	0.398
4.99	0.716	24.86	2.46	49.09	0.488
5.99	0.680	25.85	2.38	49.60	0.558
6.99	0.640	26.85	2.34	50.10	0.608
7.98	0.594	27.85	2.30	50.60	0.638
8.98	0.542	28.85	2.24	51.12	0.654
9.98	0.486	29.85	2.18	51.63	0.658
10.98	0.426	30.84	2.10	52.14	0.650
11.98	0.362	31.84	2.04	52.64	0.636
12.97	0.296	32.84	1.946	53.15	0.614
13.97	0.228	33.84	1.864	53.66	0.590
14.21	0.180	34.84	1.790	54.17	0.560
14.45	0.130	35.83	1.730	54.68	0.530
14.68	0.076	36.83*	1.706	55.18	0.500
14.92	0.022	37.45	1.650	55.69	0.468
15.56	0.003	38.10	1.498	56.71	0.406
16.19	0.0028	38.64	1.262	57.72	0.348
16.83	0.0172	39.37	0.954	58.74	0.296
17.45	0.130	40.22	0.780	59.76	0.248
18.41	1.114	41.06	0.618	60.77	0.208
19.37	1.868	41.91	0.462	61.79	0.172
20.32	2.36	42.76	0.314	62.80	0.140
21.27	2.54	43.60	0.172	63.82	0.112
21.35	2.54	44.45	0.038	64.84	0.088
21.43	2.54	44.61	0.030	65.85	0.066
21.51	2.56	44.77	0.028	66.87	0.048
21.59	2.56	44.93	0.028	67.88	0.030
21.90	2.68	45.08	0.030	68.90	0.012

*Boundaries of main fuel annulus.

Table C.3. Gold Foil Measurements of Thermal Neutron Fluxes in TSR-II Core.
Separation between ^{235}U-loaded cover plates and B_4C control plates = 2.16 cm.

Radius (cm)	Thermal-Neutron Flux (neutrons·cm^{-2}·sec^{-1}·MW^{-1})
	x 10^{12}
22.86	2.50
23.86	2.46
24.86	2.42
25.85	2.38
26.85	2.34
27.85	2.30
28.85	2.24
29.85	2.18
30.84	2.10
31.84	2.02
32.84	1.95
33.84	1.88
34.84	1.81
35.83	1.76
36.83	1.74

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